Infrastructure and methodologies for the justification of nuclear power programmes

Related titles:

Geological repository systems for safe disposal of spent nuclear fuels and radioactive waste
(ISBN 978-1-84569-542-2)
The long term safety of spent nuclear fuel and radioactive waste materials must be assured without active human oversight, based on the requirement that we do not pass the burden of nuclear waste onto future generations. Geological disposal systems and technology have been developed to isolate nuclear wastes from the human environment. This book critically reviews state-of-the-art technologies, scientific methods and engineering practices directly related to the design, operation and safety of geological repositories.

Handbook of advanced radioactive waste conditioning technologies
(ISBN 978-1-84569-626-9)
Radioactive wastes are generated from a wide range of sources presenting a variety of challenges in dealing with a diverse set of radionuclides of varying concentrations. Conditioning technologies are essential for the encapsulation and immobilisation of these radioactive wastes, forming the initial engineered barrier required for their transportation, storage and disposal. The need to ensure the long term performance of radioactive waste forms is a key driver in the development of advanced conditioning technologies. This book provides a comprehensive and systematic reference on the various options available and as well as those under development for the treatment and immobilisation of radioactive wastes.

Advanced separation techniques for nuclear fuel reprocessing and radioactive waste treatment
(ISBN 978-1-84569-501-9)
Advanced separation technology is key to closing the nuclear fuel cycle and relieving future generations from the burden of radioactive waste produced by the nuclear power industry. Nuclear fuel reprocessing allows for reuse of useful fuel components for further power generation, while the separation of actinides, lanthanides and other fission products means that residual radioactive waste can be minimised. The future of the nuclear industry relies on the advancement of this technology to ensure environmental protection, criticality-safety and non-proliferation. This book provides a comprehensive review of the fundamental issues of radioactive materials separations.

Details of these and other Woodhead Publishing books can be obtained by:

- visiting our web site at www.woodheadpublishing.com
- contacting Customer Services (e-mail: sales@woodheadpublishing.com; fax: +44 (0) 1223 832819; tel.: +44 (0) 1223 499140 ext. 130; address: Woodhead Publishing Limited, 80 High Street, Sawston, Cambridge CB22 3HJ, UK)
- contacting our US office (e-mail: usmarketing@woodheadpublishing.com; tel.: (215) 928 9112; address: Woodhead Publishing, 1518 Walnut Street, Suite 1100, Philadelphia, PA 19102-3406, USA

If you would like to receive information on forthcoming titles, please send your address details to: Francis Dodds (address, tel. and fax as above; e-mail: francis.dodds@woodheadpublishing.com). Please confirm which subject areas you are interested in.

Woodhead Publishing Series in Energy: Number 28

Infrastructure and methodologies for the justification of nuclear power programmes

Edited by
Agustín Alonso

WOODHEAD PUBLISHING

Oxford Cambridge Philadelphia New Delhi

Published by Woodhead Publishing Limited,
80 High Street, Sawston, Cambridge CB22 3HJ, UK
www.woodheadpublishing.com

Woodhead Publishing, 1518 Walnut Street, Suite 1100, Philadelphia, PA 19102-3406, USA

Woodhead Publishing India Private Limited, G-2, Vardaan House, 7/28 Ansari Road, Daryaganj, New Delhi – 110002, India
www.woodheadpublishingindia.com

First published 2012, Woodhead Publishing Limited

British Library Cataloguing in Publication Data
A catalogue record for this book is available from the British Library.

Library of Congress Control Number: 2011943279

ISBN 978-1-84569-973-4 (print)
ISBN 978-0-85709-377-6 (online)
ISSN 2044-9364 Woodhead Publishing Series in Energy (print)
ISSN 2044-9372 Woodhead Publishing Series in Energy (online)

The publisher's policy is to use permanent paper from mills that operate a sustainable forestry policy, and which has been manufactured from pulp which is processed using acid-free and elemental chlorine-free practices. Furthermore, the publisher ensures that the text paper and cover board used have met acceptable environmental accreditation standards.

Typeset by Toppan Best-set Premedia Limited, Hong Kong
Printed by TJI Digital, Padstow, Cornwall, UK

Contents

Contributor contact details

(* = main contact)

Editor and Chapters 1, 8 and 18

A. Alonso
Nuclear Engineering Department
Universidad Politécnica de Madrid
Rafael Calvo 3 2F
28010 Madrid
Spain
E-mail: agustin.alonso@nexus5.com

Chapter 2 and Appendix 2

A. Carnino
Consultant in Safety, Management of Safety, Safety Culture and Security
73 Impasse de la Sarriette
83230 Bormes les Mimosas
France
E-mail: annick.carnino1@orange.fr

Chapter 3

D. F. Torgerson
Atomic Energy of Canada Ltd (AECL)
Chalk River Laboratories
Chalk River
Ontario
K0J 1J0
Canada
E-mail: torgersond@aecl.ca

Chapter 4

G. Caruso
Regulatory Activities Section
Division of Nuclear Installation Safety
Department of Nuclear Safety and Security
International Atomic Energy Agency
Vienna
Austria
E-mail: G.Caruso@iaea.org

Chapter 5

J. Moares
Independent Consultant
4 Billings Way
Cheltenham
Gloucestershire GL50 2RD
UK
E-mail: john.moares@btinternet.com

Chapter 6

F. J. Sánchez
Safety, Operation and Training Direction
Tecnatom
Avda. Montes de Oca, 1
San Sebastián de los Reyes
28703 Madrid
Spain
E-mail: fsanchez@tecnatom.es

Chapter 7

S. K. Sharma
Atomic Energy Regulatory Board of India
301/1, Sagar Darshan
Plot-38, Sector-18
Nerul (West)
Navi Mumbai 400706
India
E-mail: sksharma.aerb@gmail.com

Chapter 9

S. Bilbao y León*
Department of Mechanical and Nuclear Engineering
Virginia Commonwealth University
401 W. Main Street
PO Box 843015
Richmond, VA 23284-3015
USA
E-mail: SBilbao@vcu.edu

J. H. Choi, J. Cleveland, I. Khamis, A. Rao, A. Stanculescu, H. Subki and B. Tyobeka
Nuclear Power Technology Development Section
Department of Nuclear Energy
International Atomic Energy Agency
Wagramer Strasse 5
PO Box 100
1400 Vienna
Austria
E-mail: I.Khamis@iaea.org

Chapter 10

D. A. Meneley
Atomic Energy of Canada Ltd
University of Ontario Institute of Technology
2330 Fire Route 26B
Lakefield
Ontario
K0L 2H0
Canada
E-mail: mmeneley@sympatico.ca

Chapter 11

A. J. González
Autoridad Regulatoria Nuclear de Argentina (Argentine Nuclear Regulatory Authority)
Av. del Libertador 8250
C1429BNP
Ciudad de Buenos Aires
Argentina
E-mail: agonzalez@arn.gob.ar; abel_j_gonzalez@yahoo.com

Chapter 12

E. Gil López
Radiation Safety Regulator
International Atomic Energy Agency
Wagramer Strasse 5
PO Box 100
Vienna
Austria
E-mail: E.Gil-Lopez@iaea.org

Chapter 13

M. S. Pellechi
Division of Operations A
Department of Safeguards
International Atomic Energy Agency
Wagramer Strasse 5
PO Box 100
1400 Vienna
Austria
E-mail: m.pellechi@iaea.org

Chapter 14

H. Forsström
SKB International AB
Box 1091
SE-10139 Stockholm
Sweden
E-mail: hg.forsstrom@telia.com

Chapter 15

H.-H. Rogner
International Atomic Energy Agency
Wagramer Strasse 5
PO Box 100
1400 Vienna
Austria
E-mail: h.h.rogner@iaea.org

Chapter 16

F. Bazile
CEA
Nuclear Energy Division
Centre de Saclay
91191 Gif-sur-Yvette Cedex
France
E-mail: fanny.bazile@cea.fr; fannybazile@yahoo.fr

Chapter 17

I. Salter*, P. Robinson, M. Freeman and J. Jagasia
Burges Salmon LLP
One Glass Wharf
Bristol BS2 0ZX
UK
E-mail: ian.salter@burges-salmon.com; patrick.robinson@burges-salmon.com; jay.jagasia@burges-salmon.com; Michael.freeman@burges-salmon.com

Chapter 19

A. González
Empresarios Agrupados, A.I.E.
Calle Magallanes 3
28015 Madrid
Spain
E-mail: agonzalez@empre.es

Chapter 20

A. Alonso*
Nuclear Engineering Department
Universidad Politécnica de Madrid
Rafael Calvo 3 2F
28010 Madrid
Spain
E-mail: agustin.alonso@nexus5.com

S. K. Sharma
Atomic Energy Regulatory Board of India
301/1, Sagar Darshan
Plot-38, Sector-18
Nerul (West)
Navi Mumbai 400706
India
E-mail: sksharma.aerb@gmail.com

D. F. Torgerson
Atomic Energy of Canada Ltd (AECL)
Chalk River Laboratories
Chalk River
Ontario
K0J 1J0
Canada
E-mail: torgersond@aecl.ca

Chapter 21

R. Gasca
Asociación Nuclear Ascó – Vandellós II, a.i.e.
43890 L'Hospitalet de l'Infant (Tarragona)
Spain
E-mail: rgasca@anacnv.com

Chapter 22

E. Grauf
se-engineering GmbH
Panoramastrasse 7
74388 Talheim
Germany
E-mail: safe@se-grauf.de

Chapter 23

M. Lipár
Operational Safety Section
Department of Nuclear Safety and Security
International Atomic Energy Agency
Wagramer Strasse 5
PO Box 100
1400 Vienna
Austria
E-mail: M.Lipar@iaea.org

Chapter 24

T. S. LaGuardia
LaGuardia & Associates, LLC
38 Pell Mell Drive
Bethel, CT 06801
USA
E-mail: tsl8@aol.com

Appendix 1

W. E. A. Wilson
Burges Salmon LLP
One Glass Wharf
Bristol BS2 0ZX
UK
E-mail: William.Wilson@burges-salmon.com

Appendix 3

M. J. Moracho Ramirez
Division of Nuclear Installation Safety (NSNI)
International Atomic Energy Agency
Vienna
Austria
E-mail: mariamoracho@aim.com

Appendix 4

E. Lindauer
Dieselstrasse 9
51103 Köln
Germany
E-mail: erwin.lindauer@freenet.de

Appendix 5

J. Reig
Nuclear Safety
OECD/Nuclear Energy Agency
Le Seine Saint-Germain
12, boulevard des Îles
92130 Issy-les-Moulineaux
France
E-mail: javier.reig@oecd.org

Woodhead Publishing Series in Energy

1 **Generating power at high efficiency: Combined cycle technology for sustainable energy production**
Eric Jeffs

2 **Advanced separation techniques for nuclear fuel reprocessing and radioactive waste treatment**
Edited by Kenneth L. Nash and Gregg J. Lumetta

3 **Bioalcohol production: Biochemical conversion of lignocellulosic biomass**
Edited by K.W. Waldron

4 **Understanding and mitigating ageing in nuclear power plants: Materials and operational aspects of plant life management (PLiM)**
Edited by Philip G. Tipping

5 **Advanced power plant materials, design and technology**
Edited by Dermot Roddy

6 **Stand-alone and hybrid wind energy systems: Technology, energy storage and applications**
Edited by J.K. Kaldellis

7 **Biodiesel science and technology: From soil to oil**
Jan C.J. Bart, Natale Palmeri and Stefano Cavallaro

8 **Developments and innovation in carbon dioxide (CO_2) capture and storage technology Volume 1: Carbon dioxide (CO_2) capture, transport and industrial applications**
Edited by M. Mercedes Maroto-Valer

9 **Geological repository systems for safe disposal of spent nuclear fuels and radioactive waste**
Edited by Joonhong Ahn and Michael J. Apted

24 **Direct-drive renewable energy systems**
Edited by Markus Mueller and Henk Polinder

25 **Advanced membrane science and technology for sustainable energy and environmental applications**
Edited by Angelo Basile and Suzana Pereira Nunes

26 **Irradiation embrittlement of reactor pressure vessels (RPVs) in nuclear power plants**
Edited by Naoki Soneda

27 **High temperature superconductors (HTS) for energy applications**
Edited by Ziad Melhem

28 **Infrastructure and methodologies for the justification of nuclear power programmes**
Edited by Agustín Alonso

29 **Waste to energy (WtE) conversion technology**
Edited by Marco Castaldi

30 **Polymer electrolyte membrane and direct methanol fuel cell technology Volume 1: Fundamentals and performance**
Edited by Christoph Hartnig and Christina Roth

31 **Polymer electrolyte membrane and direct methanol fuel cell technology Volume 2: In situ characterisation techniques**
Edited by Christoph Hartnig and Christina Roth

32 **Combined cycle systems for near-zero emission power generation**
Edited by Ashok Rao

33 **Modern earth buildings: Materials, engineering, construction and applications**
Edited by Matthew R. Hall, Rick Lindsay and Meror Krayenhoff

34 **Handbook of metropolitan sustainability: Understanding and improving the urban environment**
Edited by Frank Zeman

35 **Functional materials for energy applications**
Edited by John Kilner, Stephen Skinner, Stuart Irvine and Peter Edwards

36 **Nuclear decommissioning: Planning, execution and international experience**
Edited by Michele Laraia

37 **Nuclear fuel cycle science and engineering**
Edited by Ian Crossland

38 **Electricity transmission, distribution and storage systems**
Edited by Ziad Melhem

39 **Advances in biodiesel preparation: Second generation processes and technologies**
Edited by Rafael Luque and Juan Antonio Melero

40 **Biomass combustion science, technology and engineering**
Edited by Lasse Rosendahl

41 **Ultra-supercritical coal power plant: Materials, technologies and optimisation**
Edited by Dongke Zhang

42 **Radionuclide behaviour in the natural environment: Science, impacts and lessons for the nuclear industry**
Edited by Horst Geckeis and Christophe Poinssot

43 **Calcium and chemical looping technology for power generation and carbon dioxide (CO_2) capture: Solid oxygen- and CO_2-carriers**
P. Fennell and E.J. Anthony

44 **Materials ageing and degradation in light water reactors: Mechanisms, modelling and mitigation**
Edited by K.L. Murty

45 **Structural alloys for power plants: Operational challenges and high-temperature materials**
Edited by Amir Shirzadi, Rob Wallach and Susan Jackson

46 **Biolubricants: Science and technology**
Jan C.J. Bart, Emanuele Gucciardi and Stefano Cavallaro

47 **Wind turbine blade design and materials: Improving reliability, cost and performance**
Edited by Povl Brøndsted and Rogier Nijssen

48 **Radioactive waste management and contaminated site clean-up: Processes, technologies and international experience**
Edited by William E. Lee, Michael I. Ojovan, Carol M. Jantzen

49 **Probabilistic methods of strength reliability and their application for optimum nuclear power plant life management (PLiM)**
Gennadij V. Arkadov, Alexander F. Getman and Anderi N. Rodionov

50 **Coal utilization in industry: Towards cleaner production**
Edited by D.G. Osborne

51 **Coal power plant materials and life assessment: Developments and applications**
Edited by Ahmed Shibli

2

The lifecycle of a nuclear power plant

A. CARNINO, Consultant in Safety, Management of Safety, Safety Culture and Security, France

Abstract: This chapter deals with the nuclear fuel cycle and briefly reviews all its phases. Nuclear power plants are then considered through their lifecycle and the development of new power plants is examined from the time of decision to embark on a nuclear programme to the decommissioning phase.

Key words: nuclear fuel cycle, nuclear power plant lifecycle, safety principles, requirements for new NPPs, international conventions, site selection, design, construction, commissioning, operation, transportation, security, decommissioning and dismantling, nuclear waste management.

2.1 Introduction

For countries deciding to launch or rebuild a nuclear power programme it is important to understand that it is a commitment of some 100 years which also requires the accomplishment of a number of steps before the final construction decision is taken. Considerations need to be given to the knowledge of the whole fuel cycle, its constraints and its requirements. In addition, economic factors linked to loan availability as well as the means to implement good safety procedures constitute prerequisites.

2.2 Overview of the complete nuclear fuel cycle

The different steps of the fuel cycle are as follows:

1. Mining and milling to extract the ore.
2. Conversion factories to extract the uranium-235 (U^{235}) from the ore and transform it into the well-known yellowcake.
3. Enrichment facilities to transform the yellowcake into UO_2 enriched to 3–4% for later production of the fuel for nuclear power plants.
4. Fuel fabrication factories.
5. Irradiation of the fuel in nuclear power plants up to the burn-up desired.
6. Irradiated fuel intermediate storage waiting for a certain decay before being sent to irradiated fuel treatment.
7. Irradiated fuel treatment with a choice between two options: either direct storage of the irradiated fuel in special packages for intermediate

Published by Woodhead Publishing Limited, 2012

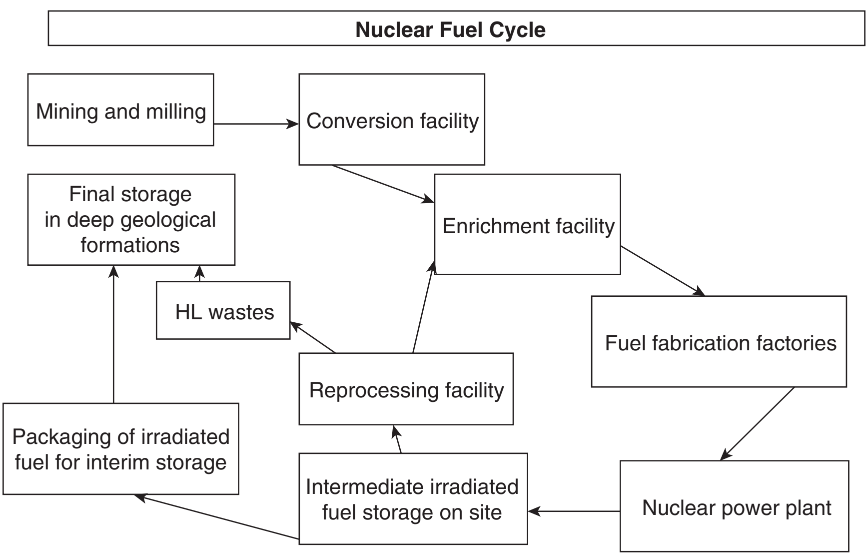
and later final storage in deep geological storage, or reprocessing for separating the plutonium and used uranium in order to reuse them in mixed oxide fuel (called MOX), and the rest being compacted and vitrified into wastes containers for final storage.

8. Final storage in deep geological formations.

These are shown together with their interconnections in Fig. 2.1. The steps up to loading the fuel in a nuclear power plant are called the front end of the fuel cycle, and the steps after unloading the fuel from the reactor are called the back end.

For accomplishing all these steps transportation of radioactive materials and fuel is necessary. The main challenges in the whole fuel cycle are proliferation resistance, security and safety as well as ensuring the sustainability of uranium and fuel supply, site remediation after closure of the factories and final siting for disposal of wastes.

The total reported uranium resources in the world in 2009 were 5400/6300 th.tU (reasonably assured resources/inferred resources). These would last for 100 years at recent demand level (source: IAEA). The fuel cost is an advantage for the industry. The price fluctuates depending on the market but recently prices have increased with the expectation of nuclear rebirth. Nuclear power is still economically viable even with increased prices. Since 2003, which was the year of the maximum price, the price has slowly fallen to now some 50 US dollars per pound. The remaining problem is security of supply for all countries engaging in nuclear energy. Some projects are



2.1 The nuclear fuel cycle.

advancing for creating an international fuel bank under the auspices of IAEA.

Another sensitive area for the front end of the fuel cycle is the enrichment process. The key issue is the risk of proliferation: by successive iteration, highly enriched uranium, for example, can be diverted to usage in nuclear weapons. The cost of enrichment is around 160 US dollars per SWU (separative work unit). Techniques mostly used for enrichment are gaseous diffusion and centrifuges.

The fuel itself, once in the reactor, has to be highly reliable since it constitutes the first physical barrier between the radioactive material and the primary system coolant. It is also needed to reach a high burn-up, which means it can stay in the reactor core for five or six years, thus authorizing longer periods of time between refuelling.

Coming to the back end of the fuel cycle, after the fuel has been unloaded from the reactor core, the main problem is the used fuel management. At this stage the fuel contains 1% Pu and 92.5% U. The total radiotoxicity decreases with time and depends on the material considered.

If the choice has been made of reprocessing the used fuel, the process will separate the nuclear materials reusable from the wastes. 99.9% of the nuclear materials are recovered after reprocessing and the volumes of wastes have been extensively reduced. The high-level wastes are then treated by vitrification into containers for storage.

Nuclear wastes have been categorized in terms of their activity. Table 2.1 the lists terms used together with their siting recommendations. Nuclear wastes are also produced by medical applications, industrial radioactive sources, research and research reactors and accelerators. These too are also stored according to their radioactivity in the same way and with the same precautions as the reactor wastes coming from nuclear power operation or used fuel and wastes from other nuclear activities such as reprocessing and dismantling.

Table 2.1 Nuclear waste classification and disposal options

Terms used for nuclear wastes	Most agreed disposal options
EW = exempt wastes VLLW = very low level wastes	Dilute and disperse
LLW = low level wastes L/ILW = low and intermediate level wastes	Near-surface trenches
LILW = short lived L/ILW LILW-LL = long lived and intermediate level	Engineered facilities on or near surface
ILW = intermediate level wastes	Intermediate depth caverns
HLW = high level wastes	Deep geological repositories

Storage of HLW is considered as an interim solution and final HLW repositories must be found, which means that, from the beginning and throughout the lifetime of the nuclear power plant, solutions should be considered and found for countries coming to develop a nuclear programme. Final repositories are in deep geological disposal sites. The challenges are the social acceptability and the interdisciplinary tasks for safe repository siting and operation.

2.3 Overview of the nuclear power plant lifecycle

The different phases in the life of a nuclear power plant can be summarized by the following:

- Decision to build a nuclear power plant and choice of the type of plant after the bidding process
- Site selection
- Design, construction and commissioning
- Operation including periodic safety reassessment up to the end of the lifetime
- Decommissioning and dismantling
- Spent fuel management and waste storage/repository.

In parallel the availability of fuel supply should be addressed and permanent training and retraining should be a concern for the whole duration of the life (currently 60 years) with preservation and transmission of knowledge.

Documentation should also be maintained during the life of the installation in a manner easy to refer to and especially including the regulatory decisions or licence conditions. The IAEA document INSAG-19 (IAEA, 2003b) is concerned with maintaining the design integrity of nuclear installations throughout their operating lifetime and further recommends the creation of a design authority within the licensee:

> 'The need to maintain design integrity and to preserve the necessary detailed and specialized design knowledge poses a significant challenge for the organization that has overall responsibility for the safety of a plant over its operating lifetime. This organization, namely the operating organization, will therefore need to take specific and vigorous steps to assure itself that the design knowledge is maintained appropriately. The operating organization must also assure itself that a formal and rigorous design change process exists so that the actual configuration of the plant throughout its life is consistent with changes to the design, that changes can be made with full knowledge of the original design intent, the design philosophy and of all the details of implementation of the design, and that this knowledge is maintained or improved throughout the lifetime of the plant. For the process of controlling design change, the acces-

sibility of design knowledge is not a trivial matter. The amount of data is huge, as it includes, for example, original design calculations, research results, mathematical models, commissioning test results and inspection history. Further, many design change issues can be complex.'

Safety culture, which is translated into the expression 'safety first', should be implemented during the whole lifecycle of a nuclear power plant from design to decommissioning and waste management. The concept of safety culture applies to organizations, including all levels of management, and to the individuals who should always demonstrate in their attitudes and behaviours their dedication to safety. Appendix 2 of the present book gives definitions, assessment and enhancement of safety culture in nuclear installations.

2.4 Requirements for new nuclear power plants

To deal with requirements for new nuclear power plants, two main documents should be referred to: IAEA (2007b), *Milestones in the Development of a National Infrastructure for Nuclear Power*, NG-G-3.1, and IAEA (2008a), *Nuclear Safety Infrastructure for a National Nuclear Power Programme Supported by the IAEA Fundamental Safety Principles*, INSAG-22.

2.4.1 Preliminary phase

At the level of a country and upon the decision of the government, the first preliminary phase is to prepare the manpower necessary for carrying the programme, a phase that will evolve with time over periods as long as 10 years, and to ratify all international conventions necessary to meet the necessary levels of safety, security and safeguards.

Manpower development, in addition to nuclear engineering achieved through education, theoretical training and on-the-job training, if necessary in foreign countries, should follow a very long schedule to be able to man the required structure to be put in place. Such a structure is composed of the energy producer, the regulatory body, research laboratories and technical support organizations. It is recommended to man the top levels of the structure with nationals having the power of decision and well-established responsibilities. One should not forget to prepare the manpower necessary for the maintenance of the installations.

A national structure should be set for managing the whole programme. This includes the government support ministries with a clear definition of roles and responsibilities, and the availability of financial and human resources.

The international legal framework comprises the international legal framework for nuclear safety, the international legal framework for nuclear security, and the international nuclear security initiatives.

For safety there are the *Convention on Nuclear Safety* (IAEA, 1994), the *Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management* (IAEA, 1997), the *Convention on Early Notification of a Nuclear Accident* and the *Convention on Assistance in the Case of a Nuclear Accident or Radiological Emergency* (IAEA, 1986a, 1986b) complemented by bilateral agreements between neighbouring countries, *Regulations for the Safe Transport of Radioactive Materials* (IAEA, 2009c), the *Code of Conduct on the Safety and Security of Radioactive Sources* (IAEA, 2004) and the *Code of Conduct on the Safety of Research Reactors* (IAEA, 2006a). The references for developing and implementing nuclear laws are IAEA (2003c), *Nuclear Law Handbook*, and the newer IAEA (2010a), *Nuclear Law Handbook – Implementing Legislation*.

It is necessary to implement at all stages of the life of the installation the 10 fundamental principles that sustain safety as developed in IAEA (2006b), Safety Standards Series, *Fundamental Safety Principles*, SF-1:

Principle 1: Responsibility for safety.
Principle 2: Role of government.
Principle 3: Leadership and management of safety.
Principle 4: Justification of facilities and activites.
Principle 5: Optimization of protection.
Principle 6: Limitation of risk to individuals.
Principle 7: Protection of present and future generations.
Principle 8: Prevention of accidents.
Principle 9: Emergency preparedness and response.
Principle 10: Protective actions to reduce existing or unregulated radiation.

These principles constitute the starting point of the review and revision of all safety standards series of the IAEA.

For security there are the *Convention on the Physical Protection of Nuclear Material* (IAEA, 1979), the *Convention for the Suppression of Acts of Nuclear Terrorism* (IAEA, 2005), the *Convention for the Suppression of Unlawful Acts Against the Safety of Maritime Navigation* (IMO, 1988) and UN Security Council Resolution 1540 which asks 'States to prohibit non-State actors from acquiring such weapons through adoption of laws, enforcement measures, domestic controls'. In addition a certain number of initiatives need also to be implemented: the Nuclear Threat Initiative, the IAEA security plan, and the UN Global Counter-Terrorism Strategy.

The IAEA (2003c) *Nuclear Law Handbook* and the newer IAEA (2010a) *Nuclear Law Handbook – Implementing Legislation* cover all these international aspects and give all necessary details for implementation.

For safeguards, the Non-Proliferation Treaty engages all state parties to implement export controls (not to transfer nuclear material and single-use equipment and material to non-nuclear weapons States except subject to Agency safeguards (Art. III.2)), technology transfers (to facilitate fullest possible exchange of equipment, materials and scientific and technological information for peaceful uses of nuclear energy (Art. IV.2)) and disarmament (to pursue negotiations on measures relating to cessation of the nuclear arms race and to nuclear disarmament, and on treaty on general and complete disarmament under international control (Art. VI)); the verification process is in the hands of the IAEA via Comprehensive Safeguards Agreements. The structure and content of Agreements between the Agency and States, in connection with the Treaty on the Non-Proliferation of Nuclear Weapons INFCIRC/153 (Corr.), requires from countries the establishment of a State system of accounting and control (SSAC), an information system for record keeping by operators, reporting inventories, imports, exports and production of nuclear materials, providing information on design of nuclear facilities and other locations where nuclear material is used, and giving access to facilities and other locations, involving facility design information verifications, inspections and cooperation with the Agency. Chapter 13 of the present book deals in more detail with the safeguards regime.

Finally, the country should also sign one of the Conventions on Civil Liability for nuclear damage. Special arrangements have been adopted under both national laws and international legal instruments to deal with the problem of how to compensate persons for injuries and other damage that could result from nuclear incidents. Reference IAEA (2010a), *Nuclear Law Handbook – Implementing Legislation*, lists the following principles:

‘(a) A defined scope for the liability regime based on specific concepts, namely “nuclear installation”, “operator”, “nuclear incident” and “nuclear damage”
(b) Strict (no fault) liability imposed on the operator of a nuclear installation (also referred to as “absolute” liability)
(c) Exclusive liability of the operator (so-called legal channelling of liability onto one person – namely, the operator – to the exclusion of other persons)
(d) Exonerations of the operator from liability only in certain exhaustively enumerated circumstances (e.g. nuclear incidents directly due to warlike events, grave natural disasters of an exceptional character, conduct on the

part of the person suffering the damage which amounts to gross negligence or intent to cause damage)

(e) Possibility of limiting the liability in amount
(f) Mandatory financial security of the operator to meet liability
(g) Limitation of liability in time
(h) Non-discrimination and equal treatment of victims
(i) Exclusive jurisdiction of a single competent court
(j) Obligation to recognize and enforce final judgements entered by the competent court in other contracting States without re-examination of the merits.'

These Conventions will also determine the responsibilities with the vendor and constructor of the plant.

The development of the national legal framework comes next. Again the reference IAEA (2010a), *Nuclear Law Handbook – Implementing Legislation* guides countries in developing their national nuclear legal framework.

The national nuclear law needs to be established and should meet all the obligations induced by the ratification process underlined above. It should also meet the constraints of the legislative system of the country. The definition of the national nuclear law is the following: 'Nuclear law is the body of special legal norms created to regulate the conduct of legal or natural persons engaged in activities related to fissionable materials and ionizing radiation' (IAEA, 2010a).

The government should include public participation in the preliminary steps: the support of opinion support is essential for carrying out a nuclear programme. Regular information meetings could be envisaged, especially at the various steps described in the next paragraphs.

2.4.2 Site selection

The life of a nuclear power plant starts with the selection of the site. This study is necessary prior to the bidding process. The reference IAEA (2003a), Safety Standards Series, *Site Evaluation for Nuclear Installations. Safety Requirements*, NS-R-3, gives all details for proceeding to site evaluation:

'The main objective in site evaluation for nuclear installations in terms of nuclear safety is to protect the public and the environment from the radiological consequences of radioactive releases due to accidents. Releases due to normal operation should also be considered. In the evaluation of the suitability of a site for a nuclear installation, the following aspects shall be considered:

(a) The effects of external events occurring in the region of the particular site (these events could be of natural origin or human induced). Site characteristics that may affect the safety of the nuclear installation shall be investigated and assessed. Characteristics of the natural environment in the region that may be affected by potential radiological impacts in operational states and accident conditions shall be investigated. All these characteristics shall be observed and monitored throughout the lifetime of the installation.

(b) The characteristics of the site and its environment that could influence the transfer to persons and the environment of radioactive material that has been released.

(c) The population density and population distribution and other characteristics of the external zone in so far as they may affect the possibility of implementing emergency measures and the need to evaluate the risks to individuals and the population.'

If the site evaluation for the three aspects cited indicates that the site is unacceptable and the deficiencies cannot be compensated for by means of design features, measures for site protection or administrative procedures, the site shall be deemed unsuitable.

Proposed sites for nuclear installations shall be examined with regard to the frequency and severity of external natural and human-induced events and phenomena that could affect the safety of the installation. In addition to providing the technical basis for the safety analysis report to be submitted to the nuclear regulatory body, the technical information obtained for use in complying with these safety requirements will also be useful in fulfilling the requirements for the environmental impact assessment for radiological hazards. Site selection is discussed in more detail in Chapter 18 of the present book.

In addition, one should not forget to examine the electrical grid potential to receive the electricity production from the plant and make sure of its adequacy or plan the necessary modifications. This is discussed in more detail in reference IAEA (2007b), *Milestones in the Development of a National Infrastructure for Nuclear Power*, NG-G-3.1.

2.4.3 Call for bids and bid evaluation

Having determined the suitable sites, having in place the necessary infrastructure and legal instruments, the next step is to decide which vendors and plant types and power are needed for the energy plan of the country. The plan should include the forecast of energy demand over at least six decades. A small investment is necessary during this phase: the preparation of the required and available competence, the national energy requirements and the availability of the necessary funds including loans as necessary. The evaluation of the industrial capabilities of the country is to be determined at this stage since it influences the choices and decisions for the bidding process, for example a turnkey contract, or the inclusion of supplementary training for staff.

For the bid evaluation, a team should be constituted that will work with a given set of criteria depending on the specifications of the bid. Consultation with the established regulatory body is also necessary to make sure that the projects submitted meet all the safety requirements and would be licensable. It is recommended to verify that the reactor proposed in the bids is licensable in the country of the vendor and that a prototype has already been built. Depending on the bid specification, it should also be examined whether the proposal includes the necessary transfer of information supporting the construction and operation of the nuclear power plant during its lifetime. Decommissioning provisions should also be part of the project to facilitate the end of life of the installation. See Chapter 24 of the present book for detailed information on decommissioning.

The technology to choose depends on the energy plan and financial possibilities. The most advanced technologies at the present time are light water reactors, pressurized or boiling. The fuel supply guarantees need to be considered together with the spent fuel storage and final disposal.

2.4.4 Design, construction and commissioning

This phase is extremely important since it constitutes the opportunity for the national organizations such as the regulatory body and the operator to start having 'hands on' the installation. A site permit has to be issued by the regulator for starting the construction work.

The design is finalized taking into account the site characteristics. Components are ordered by the constructor. Quality assurance is the main objective to be pursued; it includes visits to the companies delivering the large components. Chapter 21 of this book gives the necessary developments for quality assurance.

The detailed instrumentation and control systems are then defined and will influence the procedures to be implemented in operation for normal functioning, incident and accident management. The reference IAEA (2009a), Safety Standards Series, *Safety Requirements for the Design of a Nuclear Power Plant*, NS-R-1, covers all aspects of designing instrumentation and control systems.

During the construction phase, the role of the national organizations is essentially to ensure the quality of all the materials and components used. To this aim, the national regulator has to perform a number of inspections. Participation and observation of the various tests during the construction require the presence of both the regulator and the operator. This will result in delivering the first authorization which is to start the commissioning phase.

The commissioning phase includes numerous functional tests. The success in this leads to the delivery of fresh fuel and fuel loading. Then starts the start-up test with different phases to obtain first criticality and later power increases up to normal power. This phase usually lasts one or two years depending on the test results. Before full power can be reached, the regulator has to issue the official permit to fuel loading and start-up. At this stage procedures for normal operation and incidents have to be available with operators trained on them with the use of functional simulators if possible. During the time of slow power increases, it is necessary to develop, test on simulator and train operators on accident procedures and accident management. Chapter 22 of this books deals with this aspect of plant commissioning.

The emergency plan should also be ready to complete the operational procedures. As referred to in the present chapter under Section 2.4.1 Preliminary phase, principle 9 of the safety fundamentals emphasizes the importance of preparing emergency planning. The necessary infrastructure may include participation of other governmental institutions, various ministries and neighbouring countries in addition to the national accident management and preparedness. The IAEA or other countries may provide assistance for implementing the emergency plans.

2.4.5 Operation

The management and organization of plant operations should be such as to ensure a high level of performance and safety in operations. Operation of the power plant is performed by monitoring and controlling the plant systems in accordance with relevant rules, operating procedures, established operational limits and conditions and administrative procedures. Shift personnel need to be authorized or licensed, both at the beginning of operation and also at regular intervals (every year, for instance). Such authorization or licensing has to be approved by the regulator. The reference IAEA (2000), Safety Standards Series, *Safety Requirements for a Nuclear Power Plant Operation*, NS-R-2, explains all the recommendations for the safe operation of a nuclear power plant.

The plant organization should clearly define the roles and responsibilities of all plant staff: the operations manager has the overall responsibility for establishing and implementing the operations programme and has the responsibility for the day-to-day running of the operations. The resources, both human and financial, need to be sufficient for all operating functions. Control room operators have to be licensed and to know all procedures and be regularly trained appropriately.

The plant organization has to fix as its first priority 'safety first' for all personnel. This means enforcing a good safety culture. Adequate training needs to be given on safety culture.

Special procedures have to be organized for maintenance, refuelling, shutdown activities, regular testing and outage activities.

The plant needs to promote plans for human resources training, retraining and authorization/licensing of operators. For a lifetime of 60 years for the plant, personnel needs should be anticipated for compensating for retirement and for new specialities that will be required, such as for decommissioning.

Operating experience and feedback is essential for ensuring the safety of the plant. It requires investigating all events which could or did affect the plant operational safety. The analysis should be shared with the regulatory body and with all the plant operators worldwide. The IAEA document IAEA (2008b), *Improving the International System for Operating Experience Feedback*, INSAG-23, and the Incident Reporting System of the IAEA/NEA demonstrate the necessity of exchanging information on events which occurred worldwide.

As the plant life goes, plant ageing management has to be put in place. It is recommended that a periodic safety review process is carried out every 10 years to provide reassurance that the licensing basis is still valid, taking into account cumulative ageing effects, obsolescence of equipment or materials, modifications implemented during the plant life, lessons from worldwide operating experience, results of advanced research and changes in international safety standards. The IAEA (2009b) Safety Standards Series document *Ageing Management for Nuclear Power Plants*, NS-G-2.12, gives all requirements for dealing with ageing.

Transportation of radioactive materials in safe and secure conditions will be prepared in the country and internationally as need be. Transportation may be by road, rail, sea or air. More information can be found in IAEA (2009c), Safety Standards Series, *Regulations for the Safe Transport of Radioactive Material*, TS-R-1; IAEA (2007a), *Considerations to Launch a Nuclear Power Programme*; and IAEA (2008c), Nuclear Security Series Guidelines, *Security in the Transport of Radioactive Material*, no. 9.

Security should be enforced to protect all radioactive material from diversion or terrorism as required by the relevant ratified conventions. All relevant information can be found in IAEA (2008c), Nuclear Security Series Guidelines, *Security in the Transport of Radioactive Material*, no. 9.

A very detailed and structured documentation programme needs to capture all operational data that may be of use for the decommissioning plan. From the beginning of the life of the installation, a decommissioning plan has to be approved by the regulatory body and to be updated regularly. At the end of the lifetime, the final plan has to be approved before stopping the installation and starting to unload the used fuel. Financial provisions should be accumulated throughout the life of the plant and kept at the level necessary for the decommissioning operations. These should include the used fuel management and the waste management resulting from the decommissioning phase.

2.4.6 Decommissioning, dismantling and waste management

As described above, a decommissioning plan has to be established and maintained throughout the plant's lifetime and accepted by the regulatory body of the country. Adequate funds should be available. This gives the authorization for final shutdown and the first operation is then to unload the used fuel. The IAEA (2006c) Safety Standards Series document, *Requirements for Decommissioning of Facilities Using Radioactive Material*, WS-R-5, gives the precise ways of dealing with decommissioning.

The regulator has the responsibility for establishing the various criteria for the safety and environment of the decommissioning phase, for performing regular inspections and for checking the handling of the produced wastes. It also has to produce criteria which have to be met for the end state allowing the release of the site from regulatory control; when this is not feasible the site has to continue to be controlled and adequate measures shall be taken.

Internationally the recommended strategy for dismantling of a nuclear plant is immediate dismantling. But some factors may induce delays (for example, availability of waste disposal and long-term repository, funds and competent personnel availability). In any case, the facility must at all times demonstrate its safety.

An organization for the management and implementation of decommissioning shall be established as part of the operating organization or delegated to contractors but the responsibility still lies with the operating organization. It is important at this stage to remember the Fundamental Safety Principle 1 on Safety Responsibility (see Section 2.4.1).

Decommissioning and dismantling involve:

- Radioactivity, which means protection of the workers and the environment
- Contamination, which induces use of chemicals and the necessity of containment.

Dismantling of an installation produces enormous quantities of wastes. The wastes generated must be disposed of or recycled when possible. The low-activity wastes can be disposed of in existing facilities that are usually available in most countries. For the other kinds of wastes it is important to reduce the volume as much as possible by separating the non-radioactive ones. The radioactive part (intermediate and high level) should be packaged and stored safely while waiting for availability of appropriate repositories.

If waste is stored on the site, a revised or new, separate authorization, including requirements for decommissioning, shall be issued for the facility.

2.5 Sources of further information and advice

IAEA

The IAEA is the world's centre of cooperation in the nuclear field. It was set up as the world's 'Atoms for Peace' organization in 1957 within the United Nations family. The Agency works with its Member States and multiple partners worldwide to promote safe, secure and peaceful nuclear technologies. Its website is of particular interest for the readers of this book and more detailed information can be found on safeguards, safety standards, nuclear law, international conventions and technical matters related to fuel cycle installations and nuclear power plants.

WNA

WNA's role is to support the global nuclear energy industry through expert Working Groups focused on industry goals and concerns representing the industry in such key policy forums as IAEA, ICRP and UN climate change talks, the WNA Public Information Service (the world's leading resource for facts on nuclear energy), WNN (the foremost nuclear news service), the Biennial WNA Market Report (an authoritative projection of the global nuclear fuel market), the industry's pre-eminent WNA Symposium and the World Nuclear Fuel Cycle conference, support for WNTI and Women-in-Nuclear operating the World Nuclear Communicators Network to foster best practice in public information on nuclear energy, and the WNA Nuclear Energy Index of globally traded nuclear stocks.

WNU

Under WNA, the World Nuclear University is a global partnership committed to enhancing international education and leadership in the peaceful applications of nuclear science and technology. The central elements of the WNU partnership are the global organizations of the nuclear industry – the World Nuclear Association (WNA) and the World Association of Nuclear Operators (WANO) – and the intergovernmental nuclear agencies – the International Atomic Energy Agency (IAEA) and the Nuclear Energy Agency of the Organization for Economic Co-operation and Development (OECD-NEA), and the leading institutions of nuclear learning in some 30 countries. WNU programmes are intended to complement existing institutions of nuclear learning by filling unmet educational and training needs on the international level. These programmes are designed to capitalize on the WNU's strength as a partnership that draws on support from industry, governments and academia. To date, WNU programmes have focused on building nuclear leadership and providing orientation on the main issues that affect the global nuclear industry today. As of September 2009, nearly 2000 nuclear professionals and students from over 60 countries had participated in such programmes called Summer Institute. Plans for future WNU programmes envisage widening their scope to include fostering industry and regulatory consensus on issues affecting nuclear industry operations, building policy consensus on a sound multinational framework to govern expanding nuclear commerce and power production, facilitating multinational academic cooperation, and enhancing public understanding of nuclear science and technology.

WANO

The World Association of Nuclear Operators unites every company and country in the world with an operating commercial nuclear power plant to achieve the highest possible standards of nuclear safety. The WANO Mission is to maximize the safety and reliability of nuclear power plants worldwide by working together to assess, benchmark and improve performance through mutual support, exchange of information and emulation of best practices.

OECD/NEA

The Nuclear Energy Agency (NEA) is a specialized agency within the Organization for Economic Co-operation and Development (OECD), an intergovernmental organization of industrialized countries, based in Paris, France. The mission of the NEA is to assist its member countries in maintaining and further developing, through international cooperation, the scientific, technological and legal bases required for the safe, environmentally friendly and economical use of nuclear energy for peaceful purposes. To achieve this, the NEA works as a forum for sharing information and experience and promoting international co-operation, a centre of excellence which helps member countries to pool and maintain their technical expertise, and a vehicle for facilitating policy analyses and developing consensus based on its technical work. The NEA's current membership consists of 29 countries, in Europe, North America and the Asia–Pacific region. Together they account for approximately 85% of the world's installed nuclear capacity. Nuclear power accounts for almost a quarter of the electricity produced in NEA member countries. The NEA works closely with the International

Atomic Energy Agency (IAEA) in Vienna – a specialized agency of the United Nations – and with the European Commission in Brussels. Within the OECD, there is close coordination with the International Energy Agency and the Environment Directorate, as well as contacts with other directorates, as appropriate. NEA areas of work are nuclear safety and regulation, nuclear energy development, radioactive waste management, radiological protection and public health, nuclear law and liability, nuclear science, the Data Bank, information and communication. For details see www.oecd-nea.org.

Euratom

The Euratom Treaty establishing the European Atomic Energy Community (Euratom) was initially created to coordinate the Member States' research programmes for the peaceful use of nuclear energy. The Euratom Treaty today helps to pool knowledge, infrastructure, and funding of nuclear energy. It ensures the security of atomic energy supply within the framework of a centralized monitoring system. Euratom acts in several areas connected with atomic energy, including research, the drawing-up of safety standards, and the peaceful uses of nuclear energy. One of the fundamental objectives of the Euratom Treaty is to ensure that all users in the European Union (EU) enjoy a regular and equitable supply of ores and nuclear fuels (source materials and special fissile materials). To this end, the Euratom Treaty created the Euratom Supply Agency, which has been operational since 1 June 1960. The Agency has the task of ensuring a regular and equitable supply of ores, source materials and special fissile materials in the EU. The Nuclear Illustrative Programme describes the status of the nuclear sector in the EU in 2006 and the possible developments in this sector, taking into account economic and environmental issues. ENSREG is the **E**uropean **N**uclear **S**afety **Re**gulators **G**roup. It is an independent authoritative expert body composed of senior officials from national regulatory or nuclear safety authorities from all 27 member states in the EU. ENSREG was established as the High Level Group on Nuclear Safety and Waste Management. The European Nuclear Energy Forum (ENEF) is a unique platform for a broad discussion, free of any taboos, on transparency issues as well as the opportunities and risks of nuclear energy. Founded in 2007, ENEF gathers all relevant stakeholders in the nuclear field: governments of the 27 EU Member States, European institutions including the European Parliament and the European Economic and Social Committee, nuclear industry, electricity consumers and the civil society. EU heads of state and government adopted an energy policy for Europe which does not simply aim to boost competitiveness and secure energy supply, but also aspires to save energy and promote climate-friendly energy sources. Taking into account the substantial contribution of nuclear energy to meeting these challenges, they endorsed the Commission proposal to organize a broad discussion among all relevant stakeholders on the opportunities and risks of nuclear energy.

WENRA

WENRA is a network of Chief Regulators of EU countries with nuclear power plants and Switzerland as well as of other interested European countries which have been granted observer status. The main objectives of WENRA are to develop a common approach to nuclear safety, to provide an independent capability to examine nuclear safety in applicant countries and to be a network of chief nuclear safety regulators in Europe exchanging experience and discussing significant safety issues. For details, contact: info@wenra.org.

In addition to all these organizations and institutions it is worth mentioning all the websites of national regulators and vendors which include a lot of useful information on nuclear energy.

2.6 References and further reading

IAEA (1960a), *Paris Convention on Third Party Liability in the Field of Nuclear Energy*.
IAEA (1960b), *Brussels Convention Supplementary to the Paris Convention*.
IAEA (1963a), *Vienna Convention on Civil Liability for Nuclear Damage*.
IAEA(1963b), *Treaty on the Non-Proliferation of Nuclear Weapons*.
IAEA (reprinted 1972), *The Structure and Content of Agreements Between the Agency and States Required in Connection with the Treaty on the Non-Proliferation of Nuclear Weapons*.
IAEA (1979), *Convention on the Physical Protection of Nuclear Material*.
IAEA (1986a), *Convention on Early Notification of a Nuclear Accident*.
IAEA (1986b), *Convention on Assistance in the Case of a Nuclear Accident or Radiological Emergency*.
IAEA (1994), *Convention on Nuclear Safety*.
IAEA (1997), *Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management*.
IAEA (2000), Safety Standards Series, *Safety Requirements for a Nuclear Power Plant Operation*, NS-R-2.
IAEA (2003a), Safety Standards Series, *Site Evaluation for Nuclear Installations. Safety Requirements*, Series NS-R-3.
IAEA (2003b), *Maintaining the Design Integrity of Nuclear Installations throughout their Operating Life*, INSAG-19.
IAEA (2003c), *Nuclear Law Handbook*.
IAEA (2004), *Code of Conduct on the Safety and Security of Radioactive Sources and the Supplementary Guidance on the Import and Export of Radioactive Sources*.
IAEA (2005), *Convention for the Suppression of Acts of Nuclear Terrorism*.

IAEA (2006a), *Code of Conduct on the Safety of Research Reactors*.

IAEA (2006b), Safety Standards Series, *Fundamental Safety Principles: Safety Fundamentals*, SF-1.

IAEA (2006c), Safety Standards Series, *Requirements for Decommissioning of Facilities Using Radioactive Material*, WS-R-5.

IAEA (2007a), *Considerations to Launch a Nuclear Power Programme*.

IAEA (2007b), Nuclear Energy Series, *Milestones in the Development of a National Infrastructure for Nuclear Power*, NG-G-3.1.

IAEA (2008a), *Nuclear Safety Infrastructure for a National Nuclear Power Programme Supported by the IAEA Fundamental Safety Principles*, INSAG-22.

IAEA (2008b), *Improving the International System for Operating Experience Feedback*, INSAG-23.

IAEA (2008c), Nuclear Security Series Guidelines, *Security in the Transport of Radioactive Material*, no. 9.

IAEA (2009a), Safety Standards Series, *Safety Requirements for the Design of a Nuclear Power Plant*, NS-R-1.

IAEA (2009b), Safety Standards Series, *Ageing Management for Nuclear Power Plants*, NS-G-2.12.

IAEA (2009c), Safety Standards Series, *Regulations for the Safe Transport of Radioactive Material*, TS-R-1.

IAEA (2010a), *Nuclear Law Handbook – Implementing Legislation*.

IAEA (2010b), *International Standards, Codes and Guides Series*.

IMO (1988), *Convention on Suppression of Unlawful Acts against the Safety of Maritime Navigation*, International Maritime Organization, Rome.

4
Regulatory requirements and practices in nuclear power programmes

G. CARUSO, International Atomic Energy Agency (IAEA), Austria

Abstract: Technology, people and organizations are the main contributors to safety and need to be considered properly in the regulatory framework in order to provide effective independent regulation of nuclear installations. The regulatory oversight includes the development and application of regulatory functions. The outcome of these functions provides the basis for determining the independent decision making on safety. The regulatory requirements, criteria and regulations that make up the selected regulatory approach are the basis for regulation and need to be properly integrated in the regulatory pyramid in accordance with the licensing process and with national arrangements.

Key words: licensing process, regulatory requirements, regulatory inspections, regulatory enforcement, regulatory oversight.

4.1 Introduction

Individual States establish the national policy for safety by means of different instruments, statutes and laws. The nuclear regulator is empowered to regulate and control nuclear activity; the regulator implements national policy by means of a regulatory programme and verifies compliance with national regulations. In general, the regulatory body develops strategies and promulgates regulations in the course of implementation of these national laws and policies.

The regulatory framework, as a major part of the national safety infrastructure, is considered in the national policy, taking into account all bodies and organizations involved and their assigned responsibilities. Building a

Published by Woodhead Publishing Limited, 2012

national safety infrastructure for nuclear power is a complex and multidisciplinary activity and can take over 10 years to complete.

In general, the policy for safety includes aspects such as general safety principles to protect both people and the environment, international legally binding and non-legally binding instruments, the regulatory system, the different governmental offices and organizations involved in nuclear safety, human resources and national safety research. These aspects are strongly dependent on national arrangements, the legal system, and current and future national considerations. The International Nuclear Safety Group (INSAG) has promoted a global safety regimen based on the IAEA Safety Standards and safety related Conventions (INSAG, 2007).

The fundamental objective with regard to nuclear safety is to protect people and the environment from the harmful effects of ionizing radiation. A comprehensive safety framework needs to be developed by the State, taking into account the Fundamental Safety Principles of the International Atomic Energy Agency (IAEA, 2006a) which represent the international consensus. Laws, ordinances and decrees are promulgated taking into consideration national policies, the current social and economic situation in the State and other specific circumstances that can influence the development of nuclear power. Normally, it is recommended to review and revise existing provisions to follow as close as possible the Fundamental Safety Principles.

4.2 Basic characteristics of regulatory organizations

There are several basic characteristics of regulatory organizations reflecting the nature of their regulatory activities. The two most relevant characteristics, which also distinguish their regulatory functions, are:

- regulatory independence
- interfaces with other regulators and coordination.

4.2.1 Regulatory independence

The fundamental element of a regulatory organization is the establishment of effective regulatory independence. The reason for this independence is to ensure that independent regulatory decisions can be made and regulatory enforcement actions taken without pressure from outside interests that may conflict with safety. The independence to take regulatory decisions affords credibility to the regulatory body in the view of the general public; the regulatory body must be seen to be independent of the organization that it regulates as well as independent of governmental organizations and industry groups that promote nuclear technologies.

The importance of regulatory independence is affirmed in the Convention on Nuclear Safety. Article 8.2 of the Convention requires: '. . . an effective separation between the functions of the regulatory body and those of any other body or organization concerned with the promotion or utilization of nuclear energy' (IAEA, 1994). However, this condition is necessary but not sufficient to obtain effective independence to regulate safety. There are internationally recognized elements that need to be considered, i.e. political, legislative, financial, competence, public information and international collaboration elements; these will be discussed in more detail below. A regulatory body cannot be absolutely independent of other parts of government: it must function within a national system of laws and budgets, just as other governmental bodies and private organizations must do. Nevertheless, for the regulatory body to have credibility and effectiveness, it should have effective independence in order to be able to make the necessary decisions in respect of protection of workers, the public and the environment. INSAG has addressed the importance of independence in regulatory decisions (INSAG, 2003).

1. *Political elements*. The political system needs to provide clear and effective separation of responsibilities and duties of the regulatory body and those of organizations promoting or developing nuclear technologies. The regulatory body should not be subject to political influence or pressure in taking decisions relating to safety. The regulatory body should, however, be accountable in respect of fulfilling its mission to protect people and the environment from undue radiation hazards. One way of ensuring this accountability is to establish a direct reporting line from the regulatory body to the highest levels of government. Where a regulatory body is part of an agency or organization that has responsibility for exploiting or promoting the development of nuclear technologies, there should be channels of reporting to a higher authority that has safety as one of its primary missions and to which the regulatory body is clearly accountable when resolving conflicts of interest that may arise. This accountability should not compromise the independence of the regulatory body in making specific decisions relating to safety with neutrality and objectivity.

2. *Legislative elements*. The functions and independence of the regulatory body with respect to safety should be defined in the legislative framework of the national regulatory system (i.e. in the laws or decrees relating to nuclear energy). The regulatory body should have the authority to adopt or to develop regulations relating to safety that give effect to laws enacted by the legislature. The regulatory body should also have the authority to take decisions, including decisions on enforcement actions. There should be a formal mechanism for appeal against regulatory decisions, with predefined conditions that must be met for an appeal to be considered.

3. *Financial elements.* The regulatory body needs adequate financial resources to discharge its assigned responsibilities. While it is recognized that the regulatory body is in principle subject to the same financial controls as the rest of government, the budget of the regulatory body should not be subject to review and approval by government agencies responsible for exploiting or promoting the development of nuclear technologies.

4. *Competence elements.* The regulatory body needs independent technical expertise in the areas relevant to its responsibilities for safety. The management of the regulatory body should therefore have the responsibility and authority to recruit staff with the skills and technical expertise it considers necessary to carry out the regulatory body's functions. In addition, the regulatory body should maintain an awareness of developments in safety-related technology. In order to assist it in its decision making on regulatory matters, the regulatory body should have access to external technical expertise and advice that is independent of any funding or support from operators or from the nuclear industry; to obtain this advice the regulatory body should have the authority to set up and fund independent advisory bodies to provide expert opinion and advice and to award contracts for research and development projects. In particular, an IAEA general safety guide (IAEA, 2002), suggests that the regulatory body should be able 'to obtain such documents and opinions from private or public organizations or persons as may be necessary and appropriate'.

5. *Public information elements.* One of the responsibilities of the regulatory body should be to inform the public. An IAEA general safety requirement (IAEA, 2010a) establishes that 'The regulatory body shall have the authority to communicate independently its regulatory requirements, decisions and opinions and their basis to the public.' The public will only have confidence in the safe use of nuclear technologies if regulatory processes are conducted and decisions are made openly. The governmental authorities should set up a system to allow independent experts and experts representing major stakeholders (for example, the nuclear industry, the workforce and the public) to provide their views on safety and related issues. The experts' findings should be made public.

6. *International collaboration*. The regulatory body needs the authority to liaise with the regulatory bodies of other countries and with international organizations to promote cooperation and the exchange of regulatory information.

4.2.2 Interfaces with other regulators and coordination

Regulatory activities are always interconnected and are also shared with different authorities or governmental organizations due to the complexity and the thematic areas involved in the regulatory process. The licensing activities of a nuclear power plant represent a clear example of the necessity to arrange several different activities to authorize every stage during the lifetime of a nuclear power plant. The clear identification of interfaces and the coordination of different authorities with responsibilities for safety within the regulatory framework need to be carried out from the very start of the licensing process. Once identified, and in order to avoid any omissions, undue duplication or conflicting requirements, it is necessary to make provisions for effective coordination. The extent of the coordination required among the numerous authorities and governmental organizations depends on the scope assigned to the nuclear regulator by the government. There are several mechanisms by which this coordination can be achieved, for example national agreements and memoranda of understandings. Clear responsibilities need to be established from the beginning and unavoidable overlaps have to be considered carefully and on a case-by-case basis.

One important aspect is that all collaboration mechanisms need to take into consideration the most appropriate form of communication among the authorities and governmental organizations involved and regular meetings should be held. Communication becomes crucial during the licensing process, in particular for a new nuclear power plant. Transparent and clear procedures need to be presented to the applicant or licensee to avoid any misunderstanding or confusion. The scope of this coordination process may vary significantly according to national arrangements. Key areas that need to be considered include:

- Safety of workers and the public
- Protection of the environment
- Applications of radiation in medicine, industry and research
- Emergency preparedness and response
- Management of radioactive waste (including government policy making and the strategy for the implementation of policy)
- Liability for nuclear damage (including international conventions and regulatory control)
- Nuclear security
- The state system of accounting for and control of nuclear material
- Safety in relation to water use and the consumption of food
- Land use, planning and construction
- Safety in the transport of dangerous goods, including nuclear material and radioactive material
- Mining and processing of radioactive ores
- Controls on the import and export of nuclear material and radioactive material.

4.3 Creation, authority, responsibilities and competence of the regulatory body

4.3.1 Establishing the regulatory body

The regulatory body is created and maintained by the State which provides it with the effective independence, legal authority, competence and resources necessary to fulfil its obligations with regard to the regulatory control of nuclear power plants. The State guarantees that the regulator will work solely on safety; i.e. no other responsibility is assigned to the regulator that might create a conflict of interest, or otherwise jeopardize its ability to perform the regulatory control function.

Before deciding to embark on a nuclear power programme, the State may already have a regulatory body regulating radioactive sources (industrial and medical sources) and/or smaller nuclear installations such as research facilities or reactors. In establishing the regulatory body for nuclear power plants, an informed decision should be made either to expand the existing regulatory body or to create a new regulatory body.

The regulatory body needs the legal authority to undertake the following:

- To develop safety principles and criteria, and to establish regulations and issue guidance that take into account the state of the art concerning safety and, in particular, international safety standards.
- To require the licensee to conduct a safety assessment, systematically or periodically during the nuclear power plant lifetime and provide all necessary safety-related information, including information from the licencee's suppliers, even if this information is proprietary.
- To issue, suspend or revoke licences and establish licensing conditions and enforcing requirements based on compliance with the regulatory body's function of verifying safety during the lifetime of the nuclear power plant.
- To arrange access, solely or together with the licensee party or applicant, to carry out inspections on the premises of any designer, supplier, manufacturer, constructor, contractor or operating organization associated with the licensee; this will enable the development of a regulatory transparent and open approach which facilitates communication with governmental authorities, the public, national and international organizations and regulators and also enables regulatory decisions and information on incidents and abnormal occurrences to be disseminated clearly.

The responsibility of the regulatory body is to protect people, society, the environment and future generations from the harmful effects of radiation.

Its role is to oversee the nuclear power programme to ensure that nuclear energy is safe to use. The prime responsibility for safety will be assigned to the operator. The operator is responsible for ensuring safety in the siting, design, construction, commissioning, operation, decommissioning, close-out or closure of its facilities, including, as appropriate, rehabilitation of contaminated areas; the operator is also responsible for the safety of activities in which radioactive materials are used, transported or handled.

Compliance with the requirements imposed by the regulatory body does not relieve the operator of its prime responsibility for safety. The operator needs to demonstrate to the satisfaction of the regulatory body that this responsibility has been and will continue to be discharged as established in the IAEA general safety requirements (IAEA, 2010a).

4.3.2 The regulatory body's role in ensuring competence

An essential element of the national policy and strategy for safety is to ensure the necessary professional training to maintain the competence of sufficient suitably qualified and experienced staff.

The building of competence is required for all parties with responsibilities for safety, including licensees, the regulatory body and organizations providing services or expert advice on matters relating to safety. Competence is necessary in the context of the regulatory framework for safety, and can be achieved by such means as technical training, learning through academic institutions and other learning centres, and research and development work.

The State should make adequate arrangements for the regulatory body and its support organizations to build and maintain expertise in the disciplines necessary for discharging the regulatory body's responsibilities in relation to safety. In cases where the training programmes available in the State are insufficient, arrangements for training should be made with other States or with international organizations. The development of the necessary competences for the regulatory control of nuclear power plants is facilitated by the establishment of, or participation in, centres where research and development work and practical applications are carried out in key areas for safety.

The regulatory body has to have appropriately qualified and competent staff. A human resources plan should be developed that states the number of staff necessary and the essential knowledge, skills and abilities for them to perform all the necessary regulatory functions. The human resources plan for the regulatory body covers recruitment, staff rotation and the processes to be used to obtain staff with appropriate competence and skills, and also includes a strategy to compensate for the departure of qualified staff. To develop and maintain the necessary competence and skills it is necessary to establish a process for knowledge management. This process includes the development of a specific training programme on the basis of an analysis of the required competence and skills. The training programme in general covers principles, concepts and technological aspects, as well as the procedures followed by the regulatory body for assessing applications for licensing, for inspecting and for enforcing the regulatory requirements in a nuclear power plant.

4.3.3 The regulatory body's role in assuring safety

The regulatory control has to be stable and consistent with a formal process based on specified policies, principles and associated criteria; the process should follow specified procedures as established in the management system. Stability and consistency of regulatory control need to be ensured by preventing any subjectivity in the decision making of the individual staff members of the regulatory body. The regulatory decisions need to be justified with the proper rationale to be traceable and supported. When carrying out its reviews, assessments and inspections, the regulatory body should inform the licensee of the objectives, principles and associated criteria for safety on which its requirements, judgements and decisions are based (IAEA, 2011a).

The regulatory body should always emphasize the continuous improvement of safety as a general objective. Changes in regulatory requirements are subject to careful scrutiny to evaluate the possible enhancements in safety that could be achieved. Any proposed changes in regulatory requirements need to be based on informed judgements and there should also be the opportunity for consultation with the licensee.

4.4 Development, functions and management system of the regulatory body

4.4.1 Developing the structure of the organization

The regulatory body is structured and organized in order to fulfil its responsibilities and to perform its functions effectively and efficiently. There are key elements to consider in the organization of the regulatory body.

Primarily the regulatory organization needs to take into account its regulatory functions: licensing; review and assessment; inspection; enforcement; and the development of regulations and guides. The objective of these regulatory functions is the verification and assessment of safety in compliance with regulatory requirements.

The national legal arrangements, regulatory infrastructure and policy direction given by the State represent one key element of the organization.

National legal arrangements and structure have a significant influence on the regulatory body, including the need to consider the requirements of regulatory bodies in other areas of industry (IAEA, 2010a).

The regulatory body will need to decide whether or not to utilize technical or other expert professional advice or services to assist it in discharging its responsibilities; this decision will influence the body's own organizational structure. The professional advice or services could be provided by advisory bodies, dedicated technical support organizations, consultants, other regulatory bodies or national and international agencies. However, the regulatory body should ensure that its organization has sufficient resources with the necessary competence to allow it to make effective decisions.

The regulatory body's organizational structure will also be affected by whether its staff are all located in a single central headquarters or whether some staff are regionally located. In considering whether to locate staff regionally there are a number of factors that need to be taken into account, including the type and geographical spread of the nuclear power plants, the number of inspectors and the amount of time they need to spend on site to fulfil their duties.

The scope of the regulatory activities in relation to safety, security and safeguards may be comprehensive or may be distributed in different regulatory organizations; this latter approach may be necessary because of the wide range of the activities covered by the regulatory oversight of planning, licensing and operation, i.e. construction, manufacturing of components, training and qualification, technical specifications, maintenance, surveillance testing, management of modifications, fire protection, radiation protection, emergency preparedness, and the management system of both the operating organization and the various suppliers.

As a case in point, licensing activities essential to the development of a nuclear facility site may be carried out either by very few or by a larger number of governmental authorities, depending on the structure and functions of the regulatory body as established by law. In some States, it is the practice for the regulatory body to approve the various suppliers involved, following audits and inspections of their management systems. Once the regulatory body issues the construction licence, construction starts, including the manufacture of important safety (and safety-related) systems and components. The construction should proceed in a manner that ensures quality and safe operation. In this phase, the operating organizations, and the regulatory body as applicable, should monitor continuously the construction of safety-related structures, systems and components, both at the site and at manufacturing facilities, to ensure that the construction is in accordance with the approved design.

In a regulatory organization, each of its functions may be assigned to an organizational unit with its own specialists. However, it is often practical and efficient to group the specialists in a matrix such that each organizational unit that is assigned responsibility for a particular function can draw on the necessary specialist skills. There is a particular need for interaction and integration between assessment and inspection functions.

The nuclear power planning programme needs to consider the number and type of nuclear power plants to be regulated in a timeframe. This needs to be considered sufficiently in advance in order to formulate a comprehensive regulatory plan in terms of regulatory resources so that effective regulatory oversight can be provided at all times to all nuclear plants.

In general, in the early stages, a new regulatory body should review the experience from regulatory bodies and appropriate organizations within the State (e.g. national research organization) and from other States (including international organizations) and use this to inform its initial organizational development including the minimum core organization and staffing needs. The use of an advisory body made up of experienced national and international experts should be considered to assist this process.

The regulatory body should regularly review its organization and make adjustments as necessary to take account of its operational experience, to address regulatory changes, and to address other changes in the regulatory environment or processes. Other factors to be considered include staffing and funding issues and the outcome of both internal and external audits, evaluations and peer reviews. Lessons learned from nuclear and non-nuclear experiences are elements to consider when the organization is reviewed.

4.4.2 Regulatory functions

The following list of activities, discussed in more detail in Sections 4.6 to 4.10, should be considered as the core regulatory functions:

- Development of a regulatory pyramid (Section 4.6)
- Licensing (Section 4.7)
- Verification and oversight during construction and operation (Section 4.8):
  - independent review and assessment
  - regulatory inspections
- Enforcement function (Section 4.9)
- Transparency and openness, and the relationship with the operating organization and other stakeholders (Section 4.10).

4.4.3 The management system of the regulatory body

The regulatory body has to establish and implement a management system that will enable it to achieve its safety goals; all processes within the management system must be open and transparent. The management system should also be continuously assessed and improved.

The management system of the regulatory body has three purposes. The first purpose is to ensure that the responsibilities assigned to the regulatory body are properly discharged. The second purpose is to maintain and improve the performance of the regulatory body by means of the planning, control and supervision of its safety-related activities. The third purpose is to foster and support a safety culture in the regulatory body through the development and reinforcement of leadership and good attitudes and behaviour in relation to safety on the part of individuals and teams.

The management system maintains the efficiency and effectiveness of the regulatory body in discharging its responsibilities and performing its functions. This includes the promotion of enhancements in safety, and the fulfilment of its obligations in an appropriate, timely and cost-effective manner so as to build confidence.

The management system also describes, in a coherent manner, the planned and systematic actions necessary to provide confidence that the statutory obligations placed on the regulatory body are being fulfilled. Furthermore, regulatory requirements should be considered in conjunction with the more general requirements under the management system of the regulatory body, and this helps to prevent safety from being compromised.

The regulatory process is a formal process based on specified policies, principles and associated criteria and follows specified procedures as established in the management system. The process should ensure the stability and consistency of regulatory control and should prevent subjectivity in decision making by the individual staff members of the regulatory body. The regulatory body should be able to justify its decisions if they are challenged. In connection with its reviews and assessments and its inspections, the regulatory body should inform applicants of the objectives, principles and associated criteria for safety on which its requirements, judgements and decisions are based, as described in the IAEA general safety requirements (IAEA, 2006b).

4.5 Development of the regulatory framework and approaches

A formal definition of a regulatory framework may be considered as a system of regulations and the means to enforce them, usually established by a government, to regulate an activity. A framework may also be considered as a skeleton or a work platform that is used as the basis for constructing the regulatory system; the framework considers a set of assumptions, concepts, criteria and practices that constitute the means of implementing the regulatory functions.

The legal system, regulations and the regulatory structure and approach constitute the regulatory framework. This may vary significantly from one State to another in its complexity, arrangements, criteria, culture and practices. The approaches used in States with large nuclear power programmes may differ from those in States with small nuclear power programmes. Also, the approaches in States with a nuclear power plant vendor may differ from those in States that import nuclear power plants.

The regulatory framework needs to be based on the chosen approach and there should also be the scope for development or further adjustment as the knowledge, experience and needs of the regulatory body change. The regulatory approach is used to provide the basis for the nuclear safety regulations; to provide the regulatory actions and safety decisions; and to establish the safety rationale that is clearly understood by the regulator, the licensee and other stakeholders.

Regardless of the approach chosen, the framework needs to be developed so that there are enough staff to cover all core competences necessary to understand all the relevant safety issues of the nuclear power programme. The regulatory approach also has implications for the need for external expert support for the regulatory body.

In order to select and plan the regulatory approach, the regulatory body considers the various regulatory approaches that are applied for nuclear power programmes elsewhere, taking into account the nuclear power plant size, the State's legal and industrial practices and the guidance provided in the IAEA Safety Standards.

The regulatory approach has an impact on the licensee and also indirectly on the safety of the nuclear facilities. Regardless of the approach selected, the regulator needs to provide clear requirements to the licensees, including its safety expectations; the regulator needs to be able to identify safety significant issues, the areas of expertise needed by the regulator and licensees respectively, the resources used by the regulators and the licensees, and the level of flexibility given to the licensee to fulfil requirements; the regulator also needs to achieve public credibility for the way in which safety is regulated.

The development of the regulatory framework involves maintaining a balance between prescriptive approaches and performance-oriented approaches. This balance might also depend upon the State's legal system and regulatory approach. The approach chosen will have a major influence on the resources needed by the regulatory body, therefore the various applicable approaches need to be considered in good time, and before starting the recruitment of staff due to the impact of the approach chosen on the number and qualifications of the regulatory staff required. Before the State decides which reactor technology is going to be deployed, the regulatory body has to be aware of these two main alternative regulatory approaches: a prescriptive approach with a large number of regulations, or a performance-, function- and outcome-oriented approach. Each regulatory approach has advantages and disadvantages associated with it, and there are also approaches that combine features of these two main alternatives. When a decision to construct a nuclear power plant is made, and the particular reactor technology is chosen, the regulatory body needs to select and adopt a regulatory approach that best suits the State's needs. The regulatory body should have its chosen approach approved by the government since there will be resource implications.

A prescriptive regulatory approach places a great deal of importance on the adequacy of the regulations for safety and requires detailed development. The regulations establish clear requirements and expectations for the regulatory body as well as for the operating organization, and thus can be used to promote systematic interaction between the regulatory body and other parties. The regulations could set detailed technical requirements, or could identify issues that the operating organization and its suppliers should address and present for assessment by the regulatory body. Specific technical requirements can then be taken from relevant international industrial standards (including nuclear specific standards) or industrial standards of other States, as agreed by the regulatory body in an early stage of the licensing process for nuclear power plants. Issuing detailed regulations places a high demand on the regulatory body's resources for their development and updating, which adds to the administrative burden.

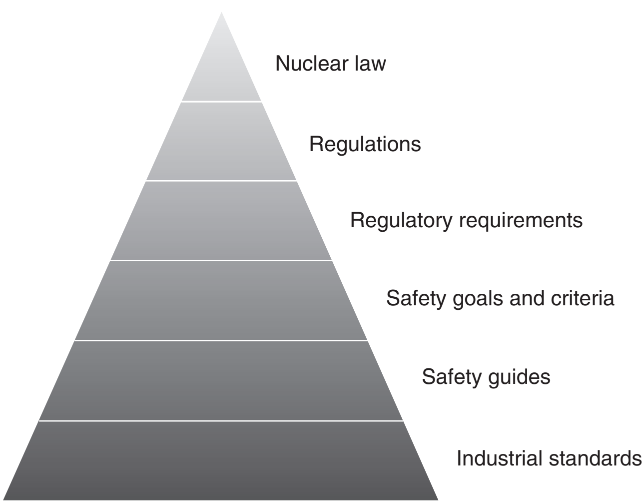
A performance-based regulatory approach allows the operating organization more flexibility in determining how to meet the established safety goals and may require fewer, less detailed regulations. However, this approach requires the establishment of specific safety goals and targets. Verifying that appropriate measures to ensure safety have been identified by the operating organization may be difficult unless the regulatory body's staff, the staff of its external support organization and the staff of the operating organization all have a high level of professional competence and are able to interact to determine whether established safety objectives for each topic are met.

Besides the general alternatives just described, the approaches in different States vary with respect to the scope and depth of safety assessment and inspection. The scope of issues that are under regulatory control may include all structures, systems and components classified as safety-relevant or may be limited to the most safety-relevant parts only. The targets of the comprehensive and systematic regulatory control and inspections are specified in a deterministic manner, on the basis of a safety classification, or they can be chosen on the basis of a probabilistic assessment of risks. As to the depth of the review, in some States the regulatory body puts the main emphasis on the assessment and auditing of the management system and the operations of the operating organizations and their suppliers. In other States the regulatory body prefers to make comprehensive independent analyses and inspections of its own. INSAG has developed the nuclear safety infrastructure necessary for a national nuclear power programme supported by the IAEA Fundamental Safety Principles (INSAG, 2008).

4.6 The regulatory function: development of a regulatory pyramid

The regulatory pyramid is the structure that represents the hierarchy of the regulatory system (Fig. 4.1). The regulations and guides are based on and suit the legal system of the State concerned and establish the principles, requirements and associated criteria for safety upon which the regulatory judgements, decisions and actions are based. The legislative and governmental mechanisms ensure that such regulations are developed and approved in accordance with appropriate time scales.

The regulatory pyramid is composed of laws and decrees, regulations, regulatory requirements, safety goals and criteria, safety guides and industrial standards. In the development of the regulatory pyramid it is always necessary to take into account the fact that the licensee or operator has the prime responsibility for safety during all phases of the nuclear power plant's life: ensuring safety in the siting, design, construction, commissioning,



4.1 The regulatory pyramid.

operation and decommissioning phases until the operator is released from the regulatory control of the plant. Therefore, it is important that the regulator clearly states that compliance with the conditions, regulations or requirements imposed by the regulatory body does not relieve the operator of its prime responsibility for safety. The licensee needs to demonstrate to the satisfaction of the regulatory body that this responsibility has been and will continue to be discharged.

4.6.1 Laws and decrees

The regulatory pyramid is established within the legal framework, using laws and decrees to establish, adopt, promote or amend regulations and guides. All regulations of the regulatory system have to be in compliance with the State legal system. In fulfilling its statutory obligations, the regulatory body establishes, promotes or adopts regulations and guides upon which its regulatory actions are based.

4.6.2 Regulations

The main purpose of regulations is to establish requirements with which all licensees must comply. In order to carry out the regulatory functions it is necessary to create a consistent and comprehensive set of regulations with adequate coverage commensurate with the associated radiation risks. The regulatory body notifies all interested parties and the public of the principles and associated criteria for safety established in its regulations and guides, and makes the regulations and guides readily available.

The development of regulations involves consultation with all interested parties and also takes into account internationally agreed standards, such as the IAEA Safety Standards, and the feedback of relevant experience from national or international organizations.

In several cases it is necessary to clearly identify by the regulatory body the reference and criteria to be used for assessing compliance with the regulations. The criteria for compliance need to be understood by the licensee from the beginning in order that it can carry out its activities correctly.

The production of regulations is included in the regulatory body's management system together with the process for establishing, revising and revoking regulations and guides. This process is established in accordance with the national legal system. The periodic review of regulations is also specified; changes need to be discussed with interested parties as appropriate but it must keep in mind that frequent changes in regulations can affect the stability or predictability of the regulatory system if the changes are not properly justified from a safety perspective.

4.6.3 Regulatory requirements

The regulations provide a framework for more detailed requirements or conditions that will be issued by the regulatory body during its activities. Requirements are also incorporated into licences as specific conditions or requirements as well as into individual authorizations or applications for licences within the licensing process. The regulatory body issues requirements when necessary to improve safety. Requirements are also necessary to improve safety considering the day-to-day regulatory oversight.

4.6.4 Safety goals and criteria

Safety goals are usually included in the regulations in compliance with the regulatory approach. In general, they are derived from the safety pillars of the IAEA Safety Standards, in particular the Fundamental Safety Principles. Safety goals may establish quantitative or qualitative criteria. The safety goals for existing reactors may also be applied for new reactors. However, it is important to consider further safety improvements that could be made at the design stage of new reactors so that safety is continuously enhanced.

There is no international consensus regarding quantitative safety goals but the current trend is to have a better balance between the deterministic and probabilistic approaches. However, some regional common approaches are useful in order to reach a global consensus at least in the qualitative way. Improved probabilistic calculations or use of operating experience to define risk magnitudes, on the one hand, and improvement in the evaluation of the consequences in terms of core damage frequency, individual doses or large release magnitudes on the other, provide the necessary relationship to establish quantitative goals.

Qualitative goals using a technology-neutral approach (meaning safety concepts and criteria independent of the type of reactor technology) can be found in the IAEA Safety Standards.

In general there are two approaches to the preparation of regulations: a prescriptive approach or a performance-based approach. The degree of application of either approach in the national regulations depends on the regulatory approach selected when establishing a regulatory framework. However, the development of regulations needs a balance between flexibility (to permit easy adaptation of the regulation to changes in circumstances and/or technology) and the need to include detailed requirements (to facilitate determination of whether the requirements have been met).

Performance-based regulations primarily specify the safety goals to be achieved rather than prescribe detailed or specific requirements. This means that the way in which the licensee is to meet the regulations is not specified by the regulator. The use of safety goals promotes the continuous safety improvement concept and provides enough flexibility to the licensee for them to determine and apply better approaches to enhance safety. This kind of regulation will not need to be changed as frequently to reflect advances in science and technology. The correct interpretation by the licensee of this type of regulation is essential; therefore it is necessary to elaborate regulatory guides in some cases to provide additional support. The verification of compliance with this type of regulation requires a high level of expertise.

With prescriptive regulations, the regulator states how safety is to be achieved with clearly defined provisions for each safety-related aspect. These provisions include the means and methods to be used in order to comply with regulatory requirements for achieving an adequate level of safety. In some cases it is easier to verify compliance with this type of regulation, but high levels of expertise are necessary for their development.

In summary, a modern regulatory system needs to include both types of regulations, to achieve the appropriate balance between performance-based and prescriptive regulations that takes into account the workload and the skills of the regulatory body's staff.

4.6.5 Safety guides

Depending on the regulatory system, a guide may or may not be mandatory; it may simply demonstrate how a certain requirement can be achieved. In some regulatory systems, the licensees can take advantage of the regulatory guidance but they may use alternative ways to demonstrate the achievements of the goals established in the requirements. These guides also provide information on data and methods to be used in assessing the adequacy of the design and on analyses and documentation to be submitted to the regulatory body by the licensee. Technological advances, research and development work, relevant operational lessons learned, and institutional knowledge can be valuable and useful when revising the guides. The management system also reflects clearly the approach to be used to review and revise guides.

4.6.6 Industrial standards and guidelines

The regulatory body also bases its regulations and guides on national legislation and utilizes existing national regulations or industrial standards (e.g. ASME Code) in areas relating to or adaptable to nuclear power plants as its initial sources of information.

4.6.7 General considerations

On a case-by-case basis, it may be beneficial to accept the use of technical standards of the vendor State or of a State having oversight experience with a reactor of the type selected. It is also useful to learn from the earlier independent analysis and safety assessments of this technology performed in other States. Furthermore, other regulatory bodies can give insights into the levels of quality achieved by key manufacturers and other suppliers, and this allows for better focusing of the auditing and evaluation of these organizations. In adapting safety standards or regulations of other States, the regulatory body should make its regulations compatible with its own national legal and regulatory framework, include appropriate requirements specific to national conditions such as special site characteristics and electrical grid conditions, promptly evaluate amendments made to the reference regulations or standards, and issue amendments to its own regulations as appropriate.

Once the vendor has been chosen through the bid evaluation process, the regulatory body should consider cooperation with the regulatory bodies of those States in which the same vendor has supplied similar plants, and especially of the State of the vendor, if possible. The possible benefits of information based on the experience of other States are clear and this could influence the tentatively planned regulatory approach.

A common option chosen in the past for regulation by States into which a first nuclear power plant was being imported was to use the regulations and standards of the supplier State. This had the advantage that the supplier knew in detail which requirements it had to meet, and it was easier for the regulatory body because it knew that such a plant was licensed in the supplier State. However, this approach has a significant disadvantage. The regulatory system of the importing State should be properly considered in the approach of the regulations adopted. If the State subsequently purchases a plant from a supplier with a different regulatory approach or a different licensing system, or if a major back-fitting programme is implemented, the two systems would have to be reconciled. The regulatory body should have a clear understanding of the basis for the regulations so that subsequent regulatory actions or changes can be fully and knowledgeably evaluated (IAEA, 2011a).

4.7 Development of the licensing process and major regulatory activities during the licensing process

A licence is a legal document issued by the regulatory body granting authorization to perform specified activities related to a nuclear power plant to a licensee who has the responsibility for safety. Chapter 20 develops nuclear power plant licensing and regulatory body and licensee related activities.

4.7.1 Development of the licensing process

The licensing process involves the granting of authorizations during all stages of the lifetime of the nuclear power plant: siting, design, construction, commissioning, operation, decommissioning and, finally, release of the site from regulatory control. This step-wise process needs to be transparent, predictable and clear, and should be in accordance with the national legal and governmental framework.

The regulator specifies the regulations, requirements and conditions for safety that are necessary during each step of the process. Compliance with these regulations, requirements and conditions is demonstrated by the licensee to the regulator, who reviews and assesses safety using clearly defined procedures. Detailed information (format and content) is specified in a time frame by the regulator in order to evaluate safety at each stage of the licensing process. The information also has to be updated regularly by the licensee, as indicated in licence conditions or regulations. The regulatory body may also need to repeat or reaffirm its assessment in order to support its decisions. The regulatory body makes decisions on the amendment, renewal, suspension or revocation of a licence based on actions such as inspections, reviews and assessments, and feedback from operational performance.

There are several types of licences: for specific time periods, for specific stages in the lifetime of the plant or for an unlimited time period. In order to grant a licence, it is necessary for a regulatory decision-making process to be in place. Political decisions are completely separate from technical decisions. These decisions are considered in a logical order, particularly when several governmental bodies are involved in the process; in this case, a licensing committee is recommended in order to integrate each governmental body into the licensing process in a timely manner. Separate hold points are specified for certain steps in the design, manufacturing, construction and commissioning processes, in order to allow verification of the results of work and to assess the preparedness to carry out the subsequent activities. The competence of licensee individuals having responsibilities for safety is verified by the regulatory body.

The early involvement of the public, in order to get public input regarding safety concerns, needs to be considered in good time by the regulator, in particular with regard to safety issues that relate to design safety requirements and the specified site conditions. Licence conditions are additional specific obligations for safety that the regulatory body considers it is necessary for the licensee to meet. In general, the conditions are incorporated into a general licence, to supplement general requirements or to make them more precise. They may include safety-related aspects affecting any stage of the plant's lifecycle: site evaluation, design, construction, commissioning, operation and decommissioning of the nuclear installation and its subsequent release from regulatory control. In general the licence conditions concern the establishment of technical limits and thresholds, specifying procedures and modes of operation, administrative matters, inspection and enforcement, and plant response to abnormal circumstances.

The licensing process should be transparent to the public, and a licence is published and made available to the public – taking into account security-sensitive and commercial proprietary information. Public participation in the licensing process gives interested parties an opportunity to present their views during certain steps of the licensing process.

There are two aspects that the regulatory body considers early in the licensing process: (a) approval of sites, and (b) certification of standardized plant designs. International cooperation also helps to facilitate the licensing process. Initially a pre-licensing stage could be adopted for an early regulatory review and approval of the proposed design safety requirements for the nuclear power plant as well as the review of the key features of the new design, to identify safety issues that would require modifications, development, or additional analysis to achieve regulatory approval of the design. During the pre-licensing process and when a particular plant design is being considered, it is necessary to provide regulatory approval of the licensee organization and site-specific aspects that may have an impact on safety in the design stage; issues that need to be adequately addressed in order to achieve regulatory approval of the site and organization should be identified.

In the combined licensing process an applicant can apply for a single licence to construct, commission and operate a nuclear installation and allow the plant to begin operation. The combined licence model requires a significant amount of regulatory resources and has only a small number of hold points, e.g. fuel loading, power increase or other technical issues.

A safety analysis report (SAR) is included in each application for a licence for a nuclear power plant. The SAR is intended to describe the facility, present the design bases and the limits on plant operation, and provide a safety analysis of the plant's structures, systems and components and of the plant as a whole. The licensing process for nuclear installations is described in an IAEA specific safety guide (IAEA, 2010b).

4.7.2 Major regulatory activities during the licensing process

Siting

The siting process for a new nuclear installation is divided into two stages: site survey and site evaluation. The government or the regulatory body identifies the potential sites and candidate sites on the basis of a set of defined criteria. Siting survey and evaluation is considered in Chapter 18.

Site evaluation is the actual selection of the site and aims to confirm the acceptability of the final site selected and to establish the parameters needed for the design of the nuclear power plant. The regulatory body should establish specific safety requirements for site evaluation, including requirements for the process of authorizing the site selected. Consequently, the regulatory body makes the review and assesses the site evaluation report, and makes a regulatory decision regarding the acceptability of the site selected and the site-related design bases.

The site evaluation process continues throughout the entire lifetime of the nuclear power plant to take into account changes in the site characteristics, in evaluation methodologies and in safety standards.

Design safety

The regulatory body establishes the nuclear safety principles and issues regulations on design; it needs to be able to evaluate the safety of the proposed design by reviewing and assessing the safety documentation (e.g. design basis, the safety analysis reports) and verifying the compliance of the design with regulatory requirements. The design basis is the range of conditions and events explicitly taken into account in the design of the nuclear installation, according to established criteria, such that the nuclear installation, through the planned operation of safety systems, can operate under these conditions and events without exceeding authorized limits. The design should be reviewed by the regulatory body considering the design basis accidents and design extended conditions. The design basis accidents (DBAs) are defined when key safety plant parameters do not exceed specified limits, with no or only minor radiological impacts, both on and off the site, and do not necessitate any off-site intervention measures, and DBAs shall be analysed in a conservative manner. The design extended conditions (DECs) consider that the plant can be brought into a controlled state, the integrity of the containment is maintained (the containment shall cope with core melt situation) and significant releases are practically eliminated. DECs may be analysed using a best-estimate approach. For those conditions that are not practically eliminated, design provisions shall be made such that only protective measures that are of limited scope in terms of area and time are necessary for the protection of the public, and sufficient time is available to implement these measures.

The IAEA established safety design requirements in 2000 (IAEA, 2000). The analysis of the operating experience and the comments by Member

States have recommended to revise the document. The revision has already been approved by the IAEA Committee of Safety Standards (CSS) but not yet published (IAEA, 2011b). Although the structure of the revised version is similar, it introduces some new concepts such as Design Extension Conditions, aimed at considering extreme circumstances.

Construction

Before construction, the main design features are assessed and approved. At this point the potential regulatory uncertainties need to be clarified. The regulatory body needs detailed construction plans, clear schedules, outlines of responsibilities of parties and information on resources required and how the licensee is to assess its own work. It is also necessary to review the way in which the licensee will promote a safety culture to its subcontractors. Significant regulatory effort is necessary during inspections to verify that new manufacturing techniques and new types of equipment meet the specifications set by the designer. The regulatory body has to review, assess and inspect, on a systematic basis, the development of the design of the installation as demonstrated in the safety documentation, in accordance with an agreed programme.

During construction, the regulatory body assesses and verifies the following: the management system of the applicant or licensee and the vendors and its subcontractors; the documentation relating to demonstration of compliance of the selected design with safety objectives and criteria, including validated results from experiments and research programmes; and the organizational and financial arrangements for decommissioning and for management of radioactive waste and spent fuel.

Commissioning

The regulatory body should conduct reviews, assessments and inspections to evaluate the commissioning test programme and the operational limits; test acceptance criteria, conditions and procedures; as-built design of the nuclear installation; non-nuclear commissioning tests; the management system; and the programme for operation. Commissioning activities are considered in detail in Chapter 22.

The results of commissioning tests should address adequately the ability of the self-assessment procedures and internal audits of the licensee to deal with deviations from design parameters. The reviews, assessments and inspections of the regulatory body assess whether the commissioning test results are adequate to confirm the adequacy of all safety-related features of the nuclear installation.

Operation

Operation is authorized only when regulatory requirements are met and the final safety analysis report has been approved, including completion of commissioning tests, recording of the results and their submission to the regulatory body for approval. Before operation, the regulator inspects, reviews and assesses the following: the results of commissioning tests; operational limits and conditions; operating instructions and procedures; and adequacy of staffing to implement these properly. Operation is considered in detail in Chapter 23.

Before and during operation the regulator has to verify the safety expectations, the management system, the operators' competence and the application of the operating experience. Any design or operational changes require significant regulatory attention.

Safety reviews should be performed on a periodic basis as requested by the regulatory body to determine the effects of ageing and to assess any plant modifications necessary to maintain safety. In general, a periodic safety review is carried out to assess the cumulative effects of plant ageing and plant modifications, operating experience, technical developments and siting aspects. The reviews include an assessment of plant design and operation against current safety standards and practices, and have the objective of ensuring a high level of safety throughout the plant's operating lifetime. They are complementary to the routine and special safety reviews and do not replace them.

Decommissioning

The regulatory body verifies compliance with the regulatory requirements of the waste management programme, spent fuel management procedures and the decommissioning programme. The regulatory body reviews, assesses, and approves, if appropriate, the final decommissioning plan and its supporting safety assessment, the management of waste and the updating of all existing safety-related documents prior to commencement of dismantling activities. Decommissioning is considered in detail in Chapter 24.

Release from regulatory control

The release of a nuclear power plant or a site from regulatory control requires, among other things, completion of decontamination and dismantling and removal of radioactive material, radioactive waste and contaminated components and structures. The regulatory body provides guidance on the radiological criteria for the removal of regulatory controls over the decommissioned nuclear installation and the site.

4.8 The compliance function: verification and oversight during construction and operation

The regulatory body oversees the compliance with regulations, regulatory requirements and the conditions specified in the licence through review and assessment and regulatory inspection. This review and assessment of information is performed prior to granting a licence and during the lifetime of the nuclear power plant. There are several types of review: pre-licensing review, review of the design or of operational changes at the plant, review of the application of the operational experience, review of the ageing of the plant, and review of licence extension or long-term operation beyond the licensing basis.

The independent review and assessment and the regulatory inspections are not separate and distinct processes. The safety verification and oversight is an integrated and interactive process that involves both the independent review and assessment processes and the regulatory inspections; from the results of both of these activities, conclusions can be drawn and therefore regulatory decisions taken. Coordination between review and assessment activities and inspection activities is a key element to ensure a systematic approach.

The results and decisions from the oversight are necessary to take appropriate regulatory actions for safety, including enforcement action when necessary. The results of reviews and assessments should be provided as feedback information for the regulatory process.

The review and assessment activities and inspection activities are the major functions carried out by the regulatory body. Therefore it is necessary to consider the human and financial resources and regulatory competences required to perform these activities.

A crucial aspect is to provide the licensee with a comprehensive understanding of national safety requirements, or references to international or other countries' requirements clearly specified as part of the licensing and regulatory framework in the early stages of the project. Vendor, licensee and regulator need to have a clear understanding of the licensing, regulatory, and inspection practices in the State where the plant is designed and in the State where the plant will be constructed.

4.8.1 Independent review and assessment

The review of the compliance with safety principles, goals and criteria, using the information provided by the licensee, is a critical task. This information needs to be accurate and should be sufficient to demonstrate the safety of the nuclear power plant considering, among other factors, the site interaction with the plant, the operational limits, test acceptance criteria for commissioning, all safety-related features of the design, safety of the operational modes and plant states, and decommissioning aspects.

All plant states (normal operation, anticipated operational occurrences and accident conditions) have to comply with the regulations and safety criteria. Therefore it is essential that the regulatory body defines and makes available to the operator the regulations and criteria and the basis for compliance that are to be applied during reviews and on which its judgements and decisions are based.

The regulatory programme for review and assessment includes all stages of the development of the nuclear power plant – from initial selection of the site, through design, construction, commissioning and operation, and including decommissioning until the plant is released from regulatory control. The review also covers deterministic and probabilistic safety analysis as needed to verify safety. A set of conservative deterministic rules and requirements for the design and operation of the plant or for the planning and conduct of activities is prepared for anticipated operational occurrences and postulated accident conditions. The probabilistic safety analysis determines all safety-significant contributors and evaluates the safety balance of the plant. One of the outputs is to verify the probabilistic safety criteria, if they have been defined. Probabilistic approaches provide insights into the reliability of the plant systems, interactions and weaknesses in the design, the application of defence in depth, and risks that it may not be possible to derive from a deterministic analysis.

The regulatory body specifies the time frame for submission of all necessary documentation, indicating the period of time that is considered necessary for the review and assessment process; this facilitates the process and minimizes delays in granting the license or any other authorization. Usually, the licensee needs to carry out additional work to complete the information for demonstration of safety. In such cases it is very beneficial to monitor the progress of documents by the operator and the progress of the review and assessment process taking into account a tentative schedule agreed by the operator and the regulatory body.

The safety analysis performed by the licensee and the results of regulatory review and assessment are major relevant inputs to define the regulatory inspection programme.

In the pre-licensing stage, it is necessary to arrange an early regulatory review and approval of the proposed design safety requirements, for the plant selected and the review of the key features of the design to identify safety issues that would require modifications, development, or additional analysis in order to achieve an initial regulatory approval.

4.8.2 Regulatory inspections

The regulatory inspections are performed to verify compliance with the regulatory requirements and with the conditions specified in the licence during all stages of the licensing process: siting, design, construction, commissioning, operation and decommissioning until release from regulatory control. These independent inspections will not relieve the licensee of its responsibility for safety.

The main purposes of regulatory inspection are to ensure that: (a) the operator is managing safety to meet, as a minimum, the safety goals, criteria and regulations established by the regulatory body; (b) the structures, systems and components in the plant meet all necessary requirements; (c) safety-related documents and instructions are valid and applied; (d) the key licensee staff have the proper competence on safety; and (e) any corrective actions resulting from operational experience are properly applied.

In order to perform the above-mentioned activities, the regulatory body will prepare a systematic inspection programme. In terms of scope and resources, this programme is planned in line with the type of regulations (prescriptive, performance-oriented or both) consistent with the regulatory approach and the way in which the regulatory body needs to verify compliance with ensuring safety at all times. In addition, the inspection efforts – scope, frequency and number of inspectors involved – have to be graded in accordance with the criticality of the different safety aspects of the plant.

It is crucial that the regulator and the licensee have a clear understanding of the inspection programme prepared for all stages of the licensing process including areas to be subject to inspection, inspection methods, selection of inspection samples and the technical information needed. Hold points need to be discussed with the licensee from the beginning to provide them with a clear understanding of the regulatory considerations that need to be taken into account.

Routine inspections may be carried out by resident inspectors or by dedicated inspectors from the regulatory headquarters, depending on the size of the nuclear programme and the geographical distribution of nuclear power plants within the State. Other inspection types, such as unannounced inspections and specific inspections (covering thematic areas or particular safety aspects), need to be part of the inspection programme.

The regulatory inspectors at the plant should have free access to the plant at any time; this is a precondition to performing the inspections properly, and these inspections are the major regulatory function to verify safety compliance.

A comprehensive inspection programme includes the regulatory inspection of the vendors, key contractors and other service providers to verify safety compliance, in particular with their quality management system –

including safety culture – and their liaison with the licensee. In addition, it may include participation in regular management meetings (for construction or operation) at the plant site and also verifying the roles and responsibilities of the licensee.

For new plants, it is also relevant that in establishing or modifying the content and schedule of an inspection programme, the regulatory body considers the results of previous inspections and the inspection experience of similar plant in another States.

In view of the significance of the safety issue, the communication of information, findings, recommendations and conclusions from regulatory inspections is planned at several levels; i.e. information needs to be communicated to the regulatory body and to other governmental bodies or interested parties.

4.9 The enforcement function

The regulatory body establishes and implements an enforcement policy within the legal framework for responding to a licensee's non-compliance with regulatory requirements or with conditions specified in the licence. In the event that risks are identified, including risks unforeseen in the authorization process, the regulatory body requires corrective actions to be taken by the licensee.

The implementation of the enforcement actions considers the appropriate levels within the organizational structure; the inspector also has the authority to carry out enforcement actions during inspections if there is an imminent likelihood of safety-significant events or when there is evidence of deterioration in the level of safety.

The range of enforcement actions starts with issuing of verbal or written notification (warnings or directives or orders); the next level is the modification, suspension, or revocation of a licence until the imposition of fines commensurate with the seriousness of the non-compliance. The range of actions that might be applied needs to be clearly understood by the regulator and the operator. However, for minor safety concerns, issues of non-compliance may be solved with a discussion between the regulator and the operator, establishing a period of time to solve the concerns and indicating the regulatory criteria involved. Clear administrative procedures and guidelines governing the use and implementation of enforcement actions are necessary.

Oral or written warnings

The enforcement action may involve warnings or directives or orders; these apply in general for deviations from or violations of regulatory requirements, or unsatisfactory safety conditions. In each case, the regulator explains the basis for each violation, deviation or unsatisfactory situation and specifies a period of time for taking corrective action. The regulator can also include technical measures such as reductions in power, pressure, temperature or other parameters, including, if necessary, temporary shutdown of the facility or administrative compensatory measures.

Modification, suspension or revocation of the authorization

In the event of repetition or serious non-compliance or safety consequences, the regulatory body modifies, suspends or revokes the licence, depending on the nature and severity of the conditions at the plant.

Fines

Fines are applied in general at the corporate level, and are imposed or recommended by the regulatory body. The administering of fines to individuals by the regulatory system is strongly discouraged in general, but may also occur. Fines on the organization rather than on individuals are preferable.

4.10 Regulatory transparency and openness, and the relationship with the operating organization and other stakeholders

Transparency and openness represent the most important elements for earning public trust and building confidence in the nuclear regulatory system. A decision to launch a nuclear power programme requires a broad acceptance by all nuclear stakeholders that such a programme is justified through a clear decision-making process. Public opinions and comments need to be considered as an input to any process that is intended to lead to a decision on launching a nuclear power programme. INSAG has considered stakeholder information and participation (INSAG, 2006).

In order to balance the need for information to be disseminated with the need to protect certain sensitive and classified information, the policy and criteria for protection of such information are established and clearly communicated to all stakeholders.

In the licensing process of a nuclear power plant it is highly beneficial for the consistency of the process to secure public involvement as early as possible, in order to obtain input from the public at the stage where all safety concerns can be adequately addressed and taken into account in the review process.

The regulatory body needs to establish and maintain effective communication mechanisms for informing stakeholders, interested parties, governmental authorities and the public regarding possible radiation risks associated with facilities and activities, safety regulatory judgements and decisions and their basis, how the nuclear regulator is performing the functions necessary to assure safety, and information on nuclear accidents, incidents and abnormal occurrences. These communication mechanisms should also provide easy access to information on safety and should create opportunities for all stakeholders to express their opinions. These communication mechanisms should also consider the make-up of audiences and their different concerns, levels of knowledge and experience. It is necessary to clarify the role of the licensee from the beginning of the nuclear programme; i.e. that the licensee has as an obligation to inform the public about the possible radiation risks associated with the plant and this obligation needs to be specified in the regulations promulgated by the regulatory body or in the licence. In order to build a successful communication programme it is essential to consider communication in the regulatory budget planning and to involve dedicated personnel with technical expertise and a talent for this discipline.

The objective and functions of the regulatory body, its independence, its technical competence, the available human resources and its neutrality have to be disseminated and proactively communicated to all stakeholders, in particular to the public and interested parties.

While maintaining its independence, the regulatory body liaises with the licensee to achieve their common objectives and to discuss safety-related issues. Mutual understanding and respect achieved through frank, open relationships will provide constructive liaison on safety-related issues.

The international participation of the regulatory body – through legally binding and non-binding international instruments (e.g. Convention on Nuclear Safety), workshops, seminars and other meetings and effective bilateral agreements among regulators to share regulatory experiences on safety – contributes to increasing the credibility of the regulator. Participation and involvement in international peer reviews designed for regulators will also strengthen the regulatory effectiveness worldwide.

4.11 Regulatory support and research

Support for the regulatory body is available in the form of technical or other expert professional advice or services as necessary to assist the body in its regulatory functions; however, this does not relieve the regulatory body of its assigned responsibilities, and independent decision making still has to be undertaken by the regulatory body. In making decisions, the regulatory body needs to consider the necessary means to assess advice provided by

Published by Woodhead Publishing Limited, 2012

advisory bodies and the information submitted by the licensee. The relationship between regulatory body competence and the extent of the technical support is a delicate balance that needs to be considered from the initial stages of the establishment of a regulatory body.

There are several approaches to receiving technical or non-technical support. For example, independent advisory bodies may provide advice on a temporary or a permanent basis, or independent expert opinions can be sought from consultants with experience in the specific field. In general the advisory bodies advise and confirm whether the regulatory body has properly addressed relevant safety issues in licensing reviews. For specific areas where expertise is not available within the regulatory body, a specific contract or services from research centres or academic institutions may be used to provide analysis of technical details and background. Finally, another approach is the establishment of a dedicated support organization working on a daily basis with the regulatory body.

The composition of advisory bodies may be derived from other government departments, regulatory bodies of other States, scientific organizations, technical experts, non-government organizations and the regulated industry. Some advisory bodies can bring broad perspectives and advice to bear on the formulation of clear, practical and balanced regulatory policy and regulations. Other more technical bodies composed of members with a range of technical skills can be established to evaluate and advise on complex technical issues.

In general, the work carried out by technical support organizations involves conducting independent confirmatory analyses or research, technical assistance on the resolution of specific regulatory issues, and the development of technical bases for safety policy and regulations. In order to develop these activities, the size, scope and responsibilities of the external support organizations need to be clearly specified.

In order to avoid any conflict of interest, as a minimum, the support provided to the regulatory body should not be provided to the licensee in the same subject area and vice versa. If this is not possible domestically, then the necessary advice or assistance may be obtained from organizations in other States or from international organizations that have no such conflicts of interest. However, in cases where a gap in expertise in a significant safety area is identified within the State, the regulatory body needs to take the appropriate steps to build the necessary competence to fill this gap, using other governmental organizations if applicable (IAEA, 2010a).

The regulatory body may need other external technical services such as personal radiation dosimetry and environmental radiation monitoring, in-service testing and inspection, maintenance of special technical equipment, and metrological activities.

Regulatory research may serve to enhance the development of knowledge, competence and ownership in nuclear science and technology. Regulatory bodies use research, when necessary, for independent analysis and in order to formulate conclusions that enable regulatory decisions to be taken. Relevant areas of research include reactor physics, thermal hydraulics, materials sciences, strength analysis and probabilistic safety assessment. National research activities need to be considered and initiated as early as possible when considering launching a nuclear power programme; these programmes may be initiated within the existing institutions or within newly created institutions. For regulatory purposes, the national research programme should be focused on areas that are vital for safety.

4.12 Sources of further information and advice

The IAEA has the statutory function of establishing or adopting standards of safety and security for the protection of health, life and property against the harmful effects of ionizing radiation in the development of peaceful uses of nuclear energy and radiation. To that end the IAEA is developing a complete and satisfactory body of safety principles, requirements and safety guides in collaboration with experts from the Member States and under the supervision of international advisory bodies. The body of safety standards comprises general and specific documents. INSAG has also created a series of relevant safety documents which can be reached through the IAEA website. The IAEA has also created an International Law Series which also includes Conventions, as listed below:

- *The Physical Protection of Nuclear Material and Nuclear Facilities,* INFCIRC/225/Rev. 4 (Corrected), IAEA, Vienna (1999); *Guidance and Considerations for the Implementation of INFCIRC/225/Rev. 4*, *The Physical Protection of Nuclear Material and Nuclear Facilities,* IAEA-TECDOC-967 Rev. 1, IAEA, Vienna (2000); and *Amendment to the Convention on the Physical Protection of Nuclear Material*, IAEA International Law Series No. 2, IAEA, Vienna (2006). (The final act of the new Convention on the Physical Protection of Nuclear Material and Nuclear Facilities was approved on 8 July 2005. See http://www.iaea.org/NewsCenter/Features/PhysicalProtection/index.html)
- *Convention on Early Notification of a Nuclear Accident*, INFCIRC/335, IAEA, Vienna (1986); and *Convention on Early Notification of a Nuclear Accident and Convention on Assistance in the Case of a Nuclear Accident or Radiological Emergency*, Legal Series No. 14, IAEA, Vienna (1987).
- *Convention on Assistance in the Case of a Nuclear Accident or Radiological Emergency*, INFCIRC/336, IAEA, Vienna (1986); and *Convention on Early Notification of a Nuclear Accident and Convention*

on Assistance in the Case of a Nuclear Accident or Radiological Emergency, Legal Series No. 14, IAEA, Vienna (1987).
- *Convention on Nuclear Safety*, INFCIRC/449, IAEA, Vienna (1994).
- *Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management*, INFCIRC/546, IAEA, Vienna (1997).

4.13 References

IAEA (1994), *Convention on Nuclear Safety*, INFCIRC/449, Vienna, IAEA.

IAEA (2000), *Safety of nuclear power plants: design*, IAEA Safety Standards Series: Requirements, NS-R-1, Vienna, IAEA.

IAEA (2002), *Organization and staffing of the regulatory body for nuclear facilities*, IAEA Safety Standards Series no. GS-G-1.1, Vienna, IAEA.

IAEA (2006a), *Fundamental Safety Principles,* IAEA Safety Standards Series no. SF-1, Vienna, IAEA.

IAEA (2006b), *The management system for facilities and activities*, IAEA Safety Standards Series: General Safety Requirements, GS-R-3, Vienna, IAEA.

IAEA (2010a), *Governmental, legal and regulatory framework for safety,* IAEA Safety Standards Series no. GS-R Part 1, Vienna, IAEA.

IAEA (2010b), *Licensing process for nuclear installations*, IAEA Safety Standards Series: Specific Safety Guide no. SSG-12, Vienna, IAEA.

IAEA (2011a), *External expert support on safety issues*, IAEA Safety Standards Series, Specific Safety Guide no. SSG-16, Standards under development, DS429, Vienna, IAEA.

IAEA (2011b), *Safety of nuclear power plants: design*, IAEA Safety Standards Series. Standards under development, DS414, Vienna, IAEA.

INSAG (2003), *Independence in regulatory decision making,* INSAG-17, Vienna, IAEA.

INSAG (2006), *Stakeholder involvement in nuclear issues*, INSAG-20, Vienna, IAEA.

INSAG (2007), *Strengthening the Global Safety Regime*, INSAG-21, Vienna, IAEA.

INSAG (2008), *Nuclear safety infrastructure for a national nuclear power programme supported by the IAEA Fundamental Safety Principles,* INSAG-22, Vienna, IAEA.

5

Responsibilities of the nuclear operator in nuclear power programmes

J. MOARES, Independent Consultant, UK

Abstract: The role of the operating organisation commences with the strategic and economic decision by the sponsoring organisation to construct a nuclear power plant. All phases of the lifecycle of a nuclear power plant are subjected to controls and regulations designed to protect the public and the workforce from any risks associated with their operation. The role of the operating organisation covers design appraisal, site appraisal and infrastructure development, the design, development and maintenance of the of the operating organisation, construction, commissioning, operations and maintenance and decommissioning. The operating lifetime will last many decades and throughout that time the organisation will need to adapt to the changing roles, and be refreshed to cater for the effects of aging in the workforce. This chapter seeks to characterise the roles at each of the phases of the life cycle and share insights into the ways in which they can be enacted.

Key words: nuclear power plant, operators, safe operation, safety responsibilities.

5.1 Introduction

Nuclear power plants (NPPs) exist in order to satisfy a public need for the safe, reliable and economic supply of electricity. NPPs must be designed, manufactured and constructed to standards that ensure that the risks associated with their operation are mitigated and the benefits to society are achieved. Governments and regulators are responsible for specifying design standards that will satisfy the safety requirements and for approving and licensing designs submitted for construction.

Regulators specify the conditions under which the plants must be operated and maintained in order to ensure compliance with the design requirements, and to safeguard the workers, the public and the environment from NPP-derived hazards. The conditions are detailed in site-specific nuclear site licences and authorisations.

The operating organisation is responsible for the safe and economic operation of the plant and for ensuring that all activities are conducted in compliance with the nuclear site licence conditions. In order to satisfy those requirements, the operating organisation must ensure it has the resources and competencies to fulfil the conditions of the site licence throughout the lifecycle of the plant.

This chapter will define the responsibilities of the operating organisation and the means of enacting them.

5.2 The responsibilities of the nuclear operator

5.2.1 Responsibility for safety

The prime responsibility for safety in nuclear power plants must rest with the person or organisation responsible for the facilities and activities that give rise to risks.

Principle 1 of the IAEA Fundamental Safety Principles (IAEA, 2006a, page 6) states: 'The responsibilities are defined in a license, granted to the licensee by the state regulators.' In summary, the licensee is responsible for ensuring that the workforce, the general public and the environment are protected from risks and hazards that might arise from the operation of nuclear facilities. The licence describes those measures that must be taken by the licensee to safeguard against those risks.

Those responsibilities cannot be delegated but work associated with the enactment of those responsibilities can be delegated or outsourced. In such circumstances the licensee must be able to demonstrate effective control over the specification, procurement and enactment of such work.

The management of risks involves maintenance of organisational effectiveness and design integrity, and compliance with operating rules and procedures.

5.2.2 Leadership and management of safety

All things that happen in industrial societies are a result of people's efforts. The bigger and more complex the task the greater number of people that will be involved and the more diverse the skill sets required. Nuclear power stations require, in their design, construction, commissioning and operation, large numbers of people with diverse skills. In order for them to be successful in what they do, they must be organised into a cohesive workforce with a clear vision of what they are seeking to achieve. The role of the leader in any organisation is to define that vision and create an organisation capable of delivering the vision.

Successful leaders are characterised by their behaviours, their honesty, integrity and competence that enable them to command respect and engender trust. The way they treat people will influence the sustainability of the organisation.

Leaders will exhibit their characteristics in both conscious and unconscious ways. The people who work for them will be looking for signs of their values, standards and expectations in everything they do and they will interpret them into their own actions and behaviours. Thus the corporate culture and in particular the safety culture in an organisation is promulgated. The culture of successful organisations will be characterised as one in which people want to do things right, want to work together to achieve shared objectives. The choice of leaders is probably the most critical decision of any organisation.

5.3 The means to enact responsibilities and enhance leadership effectiveness

5.3.1 Organisational factors

Operators in all regulatory environments must establish organisations within which the personnel have effective leadership and the knowledge, skills and attitudes required to manage nuclear assets.

The operator must establish programmes, processes and procedures through which the knowledge and skills of the workforce are effectively deployed. These must be applied to all phases of design, site selection, construction, commissioning, operation and maintenance of the plants. Ultimately, decommissioning activities will also require similar consideration.

The levels of resource and the competencies required will be determined by the strategy adopted for the implementation of the responsibilities assigned to them in the licence conditions. In many cases work will be outsourced and this will have an impact on the resource levels required by the operating organisation, but the responsibility for the enactment of the nuclear site licence will always remain with the licensee, so the ability to specify what work is required and to assess the adequacy of the work undertaken must be retained and maintained by the licensee.

5.3.2 Training and development

The operating organisation is responsible for ensuring that it is resourced with personnel who have the necessary knowledge and skills to perform the tasks assigned to them and for ensuring that those competencies are maintained throughout the lifecycle of the plant. This is achieved through induction or initial training at the time of recruitment and through continuing training during the period of employment.

Key to effective training is an objective analysis of training needs, facilities, equipment and suitably qualified trainers for the delivery of training, together with a process for the continued evaluation of training effectiveness.

In recognition of these requirements the US utilities have employed a systematic approach (SAT) to the development and delivery of technical training. This approach has contributed greatly to the enhancement of professionalism in the workforces and to performance improvements in the US NPPs.

In support of the US utilities, the National Nuclear Training Academy has established a set of objectives and criteria that describe the attributes of organisations with effective training programmes. In summary these are:

1. Training is used as a strategic tool to provide highly skilled and knowledgeable personnel for safe operation and to support performance improvement.
2. Management is committed to and accountable for developing and sustaining training programmes that meet NPP needs.
3. Initial training programmes use a systematic approach to the identification, design and delivery of training.
4. Continuing training uses a systematic approach to refresh and improve the knowledge and effectiveness of the workforce.
5. Training is conducted and uses methods and settings that are conducive to effective training. Effectiveness of training is confirmed through evaluation.
6. Evaluation methods are employed to systematically assess the effectiveness of training and to identify areas for improvement.

Each of these principles is outlined in the US National Academy for Nuclear Training standard ACAD 02-001. In that standard each of the principles or objectives is supported with a set of supporting criteria. Utility training programmes are periodically assessed against these criteria by NTA evaluators and those deemed to satisfy the criteria have their programmes accredited by an independent board of assessors. The practice has been adopted in UK and South African utilities.

5.3.3 Safety culture

Most of the arrangements for managing safety are in a very tangible form, which are easy to recognise and communicate. These consist of the site licence, policies, processes, procedures and organisational attributes, for example. To enact the work, staff will require knowledge and skills that can be objectively defined and instilled in the workforce.

Of even greater importance, however, is the need for personnel to go about their work with attitudes and behaviours which recognise the risks associated with the technology entrusted to them, that they act conservatively when making decisions that relate to safety, and that they strive to do their work to the best of their abilities at all times. Together these organisational attributes and the attitudes and behaviours are described as the safety culture of an organisation.

The term 'safety culture' was first introduced by the IAEA International Nuclear Safety Advisory Group (INSAG) in their Summary Report on the Post-Chernobyl Accident Review and subsequently published by them as IAEA Safety Series no. 75-INSAG-1. The term was later expanded and is now embodied in IAEA INSAG-4 published in 1991. The IAEA INSAG definition is:

> 'Safety culture is that assembly of characteristics and attitudes in organizations and individuals which establishes that, as an overriding priority, nuclear plant safety issues receive the attention warranted by their significance.'

The World Association of Nuclear Operators (WANO) in their Peer Review programme and the IAEA in their OSART programme conduct reviews that seek to assess the status of safety culture in the plants that they visit. The IAEA also conducts specific missions to assess safety culture in NPPs known as SCART missions (Safety Culture Assessment Review Teams).

Following the Davis Besse vessel head incident, the US Institute of Nuclear Power Operators (INPO) developed a set of principles which should exist in organisations with a strong safety culture (INPO, 2004). These were published as a guidance document for the industry. The World Association of Nuclear Operators adopted the same principles in 2006.

There are difficulties in distinguishing between national culture and safety culture in international programmes but the WANO and OSART performance objectives and criteria overcome these difficulties.

In the INPO/WANO Principles for a Strong Nuclear Safety Culture, safety culture is defined as:

> 'An organization's values and behaviors – modeled by its leaders and internalized by its members – that serve to make nuclear safety the overriding priority'

This definition, together with the defining principles, are not incompatible with the definition produced by the IAEA but emphasise the role of leaders in defining the corporate culture of an organisation. The principles are:

1. Everyone is personally responsible for nuclear safety.
2. Leaders demonstrate commitment to safety.
3. Trust permeates the organisation.
4. Decision making reflects safety first.
5. Nuclear technology is recognised as special and unique.
6. A questioning attitude is cultivated.

7. Organisational learning is embraced.
8. Nuclear safety undergoes constant examination.

These principles are further characterised in detail in the documents. US utilities have established a safety culture assessment programme based on evaluation against the principles.

5.3.4 Design integrity

In the design phase of any plant, measures are taken and design features are incorporated to ensure that the plant can satisfy stringent safety, reliability and economic criteria. The designs are subjected to rigorous analysis and, where feasible, testing to verify performance claims. Subsequently they will be further analysed by the prospective owner operators and by regulatory authorities. The process of evaluating the designs is very demanding and time consuming and necessarily so.

It follows therefore that it is equally important that all the components are manufactured and assembled in accordance with the design intent. Throughout the manufacturing and construction phase of any plant, measures must be taken to ensure that the plant and equipment complies fully with the licensed design requirements. The owner or operating organisation must be in a position to ensure that is the case before acceptance of the plant. This will require the operating organisation to establish programmes for evaluation of quality of components throughout the manufacturing and construction phases of the plant.

Throughout the operating lifetime of the plant, the operators have an obligation to ensure that the plant remains compliant with the licensed design. Maintenance inspection and testing programmes will be developed for this purpose.

Over time, the cumulative evolution of changes in plant performance and the condition of the plant due to in-service aging will need to be assessed. Periodic safety reviews are a common feature of regulation in many countries where comprehensive reappraisal of the plant status against the design and licensing criteria are undertaken. The periodicity will vary from regulator to regulator and the terms of reference and scope will vary.

Modifications or changes to operating procedures to address identified issues must be subjected to an approval process. The scope and criteria for approval of changes are usually related to the nuclear safety significance of the change in question.

These responsibilities will be placed on the operating organisation for the full lifecycle of the plant, which will be for several decades. It is vital, therefore, that every operating organisation establishes the knowledge base and capability to fulfil these functions at an early stage in the plant's lifecycle.

Typically, the body of personnel assembled to fulfil such a function is known as the 'Design authority' for the operating organisation. The IAEA has published two documents in the INSAG series that address the concept of a design authority:

- INSAG-14 (1999), *Safe management of the operating lifetimes of nuclear power plants*
- INSAG-19 (2003), *Maintaining the design integrity of nuclear installations throughout their operating life*

As the lifecycle of the plant will probably span the working lives of more than one generation of personnel, programmes must be established to retain the knowledge and capability to fulfil that function over several decades.

5.4 Responsibilities of the operator in the lifecycle of a nuclear power plant

5.4.1 Preconstruction activities

The decision to construct a nuclear power station involves the commitment of huge resources, so it is not one that can be taken without careful consideration. The decision-making process must be rigorous and comprehensive.

Initially the decision-making process involves an appraisal of the socio-economic circumstances, to determine if new generating capacity is required and to determine if a nuclear power station is an acceptable option to satisfy such a need.

If the politics and the demand for new generation are favourable the next phase will involve a decision to commit finance. The financial appraisal must include consideration for the choice of site, the design options, the funding options available and the electricity market model in which the plant will operate. These tasks will require specific knowledge and experience, which must be established in the operating organisation several years before build commences.

In addition the operating organisation must recognise and cater for the demands of the regulatory process associated with new build and acquisitions. These can be very lengthy and resource demanding, so they require a strategy and a fully resourced plan for dealing with them in order to optimise the timescales and costs involved.

The decision to build must also take into account the availability of a suitable infrastructure to manufacture, build and operate the plant. For many years the manufacturing sector was dormant due to a lack of orders.

Many of the major manufacturers survived on the basis of manufacturing replacement parts, design upgrades and service activities. Clearly such activities resulted in a reduction of staff and a loss of competencies, which have had to be renewed in a very short timescale.

The nuclear renaissance has come with a rapid surge in demand for plant and equipment, so manufacturing capacity is further stretched to capacity. Similarly, universities and other educational institutions that provided training for nuclear industry workers closed down courses and reassigned staff to other educational programmes (see Chapter 6).

The current approach to licensing of nuclear plants seeks to establish generically approved designs. However, each site chosen for construction must be assessed to determine if it can satisfy generic design requirements and to assess the environmental impact of that design on the site, through all phases of construction, commissioning and operation. These studies and the generic design characteristics will form the basis for the decision to grant licences and the conditions associated with the licences. The execution of such studies will require specialist skills.

The preconstruction activities are costly and resource intensive, as well as being quite lengthy in nature. It can be concluded from this brief description of the decision-making process that the prospective operator must commit significant resources and commence the construction of the operating organisation well before the decision to build is taken.

5.4.2 Construction and commissioning

The construction project strategy to be adopted by an operating organisation will depend on its circumstances. It must be recognised, however, that even with a turnkey contract, considerable resources will be required to ensure due diligence in the acquisition and construction phase.

For the commissioning phase the operating organisation must be sufficiently developed to acquire the knowledge and experience to be gained from commissioning activities and to be in a position to assume full responsibility for operation on completion of the contract.

In some countries the nuclear site licensing process has been developed to allow a general licence for construction and commissioning. There are, however, hold points in such licences that require the operator to meet conditions with regard to the competencies that must be established before, for example, nuclear fuel is delivered to the site or fuel is loaded into the reactor. Business plans and project plans must reflect these requirements.

Licensing issues are covered in Chapter 20 and commissioning in Chapter 22.

5.4.3 Operational phase

Safe operation is assured by effective design, acquisition and installation of plant and equipment that satisfy the design requirements. A licensing process and quality assurance measures must be established to ensure that these requirements are satisfied.

Throughout the operating life of the plant, it is the operating organisation's responsibility to ensure that the plant design fidelity is maintained and that it is operated and maintained in accordance with design requirements. Periodically regulators will require a design assessment to determine such things as the impact of in-service aging on the design characteristics and to assess if any countermeasures are required in the form of replacement components, modifications or changes to operating conditions.

Nuclear site licence conditions will define the responsibilities of the operator for ensuring safety and environmental protection.

The licensing process and the means of enacting licence requirements are discussed in Chapter 20, and operations in Chapter 23.

5.4.4 Decommissioning

The operating organisation is responsible for making full provision for the costs of decommissioning and disposal of hazardous waste. The financial provisions must be set aside throughout the operating lifetime of the plant. In addition the operator must ensure that the hazardous waste arisings are kept to an absolute minimum, that they are effectively stored until final disposal and that a full inventory of the hazardous waste is maintained. Prior to closure of an NPP, the operating organisation is required to seek a license for the activities involved.

Decommissioning is covered in detail in Chapter 24.

5.5 Importance of organisations for safe operation

The effective enactment of the roles and responsibilities associated with the construction, commissioning, operation and decommissioning of nuclear power plants depends on the effectiveness of the operating organisation. Throughout the lifecycle each phase of activity will have its own specific needs and challenges. Operating organisations must adapt and develop to meet those requirements. In addition the lifecycle of a NPP will last several decades so the challenge of maintaining organisational effectiveness over the full term of the lifecycle must not be underestimated.

Worldwide experience in peer reviews, OSART missions and event analysis demonstrate that most events and performance deficiencies identified in those programmes have their genesis in organisational and human performance factors.

Conversely, the US utilities achieved one of the most outstanding performance improvements on record. At the time of the Three Mile Island Accident in 1979, there were approximately 104 NPPs in service with an average Unit Capability Factor (UCF) of just 60%. Today, with the same number of plants in service, the figure is approximately 92%. That achievement is almost entirely due to improvements in organisational effectiveness and the ability to learn from experience.

One of the great features of that improvement is that it is well documented and readily replicable. Today organisations such as the IAEA, WANO and INPO facilitate the identification, sharing and promotion of good practices between utilities worldwide through, for example, their Technical Exchange, Peer Review, OSART, Technical Publications and Good Practices Programmes.

5.6 Building and maintaining an operations organisation

This section will describe a systematic approach to organisational design; it will also describe the reasons for and the means of monitoring and evaluating organisational effectiveness and identification of areas for improvement.

5.6.1 Systematic approach to organisational design

When designing any organisation the first requirement is to determine the purpose of the organisation:

- The business it will serve
- The functions that must be performed
- The tasks that must be performed
- The responsibilities that must be discharged
- The competencies that will be required
- The resources that will be needed.

Using the systematic approach to organisational design, it is also possible to determine the knowledge skills and attitudes required to fulfil those tasks.

In the preconstruction and construction phases the work will involve, for example:

- Design appraisal
- Financial appraisal

- Site selection
- Contract management
- Project management
- Quality assurance
- Legal and regulatory activities.

In the construction phase, industrial safety management is a very important and significant challenge. It is also necessary to have the capability of monitoring and evaluating the standard of construction, the integrity of plant installation and, in the case of turnkey contracts, establishing a project management overview.

Commissioning activities require the plant installation to be physically checked for compliance with legal and safety requirements as well as sound engineering practices prior to the test programmes to demonstrate compliance with safety and design performance characteristics. The skill sets will not be dissimilar to those required for normal operation and maintenance. In many cases these activities are carried out directly, or independently verified by members of the operating organisation that will eventually run the plant.

The construction and commissioning phases will be the subject of regulatory licence conditions in respect of staffing and competencies required (see Chapter 22 on commissioning).

The functions and activities performed in nuclear power plants are described in the WANO/INPO/IAEA Performance Objectives and Criteria (POs and Cs) used in peer reviews and OSART missions. Similar POs and Cs exist for the corporate functions associated with the management of NPPs.

For a nuclear power plant typical functions are:

- Organisation and administration
- Operations
- Maintenance
- Engineering
- Chemistry
- Radiological protection
- Emergency preparedness
- Training
- Fire protection.

In addition there are a number of what are called cross-functional areas which address such things as safety culture, industrial safety and work management. NPP organisations also need functions such as organisational administration, human resources, quality assurance and finance that are not prescribed in the WANO/INPO and IAEA POs and Cs.

In addition to determining the functional groups required, the structure has to be determined. Spans of control and the number of direct reports for leaders and managers, aggregation of synergistic groups, and the levels in the organisation must be determined. In many cases the management structure and resources associated with the management of nuclear power installations will need to be formally approved by the regulatory authorities and will be the subject of regulatory oversight for the duration of the licence.

Changes to such organisations are usually treated in a similar way to design changes for plant and equipment, including the need for regulatory approval prior to any change in some cases.

5.6.2 Building an organisation

Once the design criteria have been established, the next phase is to put the structure in place. Building an organisation is much like building a power plant. First the design has to be developed and approved; then the components, which must be suitably qualified for their application, must be acquired. The parts then have to be assembled and commissioned.

For each of the functions that have to be fulfilled in a nuclear power plant, specific jobs and tasks will have to be performed. These can be determined by conducting an analysis of each of the functions. The job and task analysis would define the knowledge and skills required for each position in the organisation.

Normally a utility would be expected to recruit personnel with generic academic qualifications and maybe a number with relevant skills and experience. A gap analysis of the skills acquired through recruitment against those determined through the job and task analysis will identify the knowledge and skills that have to be addressed through training.

When the resources needed to populate the organisation are in place, they must be put to work in a manner that tests their suitability for their assigned roles and the organisation within which they will work, in the same manner that plant and equipment is tested during commissioning.

Throughout the lifetime of the plant these skills will need to be regularly refreshed through repeat training. They will also need to be reviewed and revised from time to time on the basis of plant and personnel performance.

5.6.3 Integrated management systems

The personnel in any organisation have to be integrated into a cohesive unit in which roles and responsibilities are clearly understood. Typically these are described in the form of policies, processes and procedures in quality assurance (QA) programmes. Such programmes will include policies in respect of health, safety and welfare and environmental protection.

The great danger with quality assurance programmes is that they can become over-prescriptive; this in turn can result in them becoming impractical to use and maintain. Pragmatism in the development of the associated procedures is advisable. The degree of detail in such procedures can be determined by the nuclear safety significance of the issue to be covered and the need for detail.

Over time, quality assurance programmes have evolved from prescriptive programmes that are imposed to the more inclusive concept of Total Quality Management (TQM) programmes.

The concept of integrated management systems attempts to go further by taking into consideration cultural factors, such as safety culture, that influence and are important to the way in which an organisation operates (IAEA 2006b).

The Management System for Facilities and Activities Safety Requirements Series No. GS-R-3.

5.7 Monitoring and evaluating organisational effectiveness

Over time, organisations will experience changes to personnel and performance; if these changes go undetected they could have an adverse effect on nuclear safety and plant performance. Similarly, circumstances change and organisational needs will change as a result. It is important therefore that the effectiveness of organisations is regularly reviewed to ensure they are compliant with nuclear site licence conditions and company arrangements described in QA programmes and to identify areas for improvement.

The nuclear industry has developed many programmes for the evaluation of organisational effectiveness, some of which are described below.

5.7.1 Quality assurance audits

Typically, quality assurance audits determine whether organisations are compliant with company arrangements. Chapter 21 addresses QA in detail.

5.7.2 Peer reviews and OSARTs

Peer reviews and OSART missions are conducted by independent agencies such as INPO, WANO and the IAEA. Peer reviews involve peer-to-peer comparison of practices between utilities and plants to identify areas for improvement in organisational effectiveness and the promotion of best practices.

Assessments of situations, conditions and practices are compared against industry standards based on good practices which are in the form of performance objectives and criteria (Pos and Cs).

Feedback to the utility is in the form of areas for improvement (AFIs), often accompanied with insights into the underlying causes of the deficiencies identified.

The tangible product of a peer review or OSART mission is the mission report, but much of the benefit derived from hosting and participating in such missions is in the informal exchanges between peers.

Good practices identified in the course of such missions are captured and promoted through the industry in the form of Guidelines or Good Practices.

5.7.3 Nuclear oversight

Peer reviews and OSART missions are performed infrequently. The industry has recognised that the stimulus provided by them starts to peter out after a while and with that comes a loss of momentum for improvement.

Many organisations have now developed nuclear oversight functions that conduct reviews based on the peer review POs and Cs using a similar methodology in which the review process is maintained.

Typically, nuclear oversight personnel are independent of any line function and involve personnel with plant experience and preferably experience in the peer review process.

5.7.4 Self-assessment

Self-assessment programmes are a means by which practitioners of various programmes or functions take time out from their normal day-to-day activities to objectively assess the way in which they are conducting their activities against a set of internationally recognised criteria. Normally, WANO peer review or OSART performance objectives and criteria are used in such processes. The assessments are conducted with in-house personnel and can follow a similar format, but with limited scope, to a peer review exercise.

5.7.5 Corrective action programmes

Corrective action programmes (CAPs) are designed to enable the reporting of any conditions adverse to quality by any member of staff working at an NPP. The CAP programmes are also used to classify and trend issues, to record and monitor progress against actions raised in response to reported issues.

Corrective action programmes are also used to record and trend actions arising from other forms of evaluation such as peer reviews, Op Ex, QA audits and self-assessments. Having the corrective actions from all sources of evaluation enables the operator to ensure that common causes and contributors to issues raised are treated more efficiently and effectively.

5.8 Maintaining organisations

Over time, organisations and the individuals in them can change in both composition and performance just as the condition and performance of plant and equipment changes as a result of in-service aging. Engineers will be familiar with plant aging issues and the maintenance practices that seek to reverse or halt the impact of in-service degradation.

The same imperatives apply to organisations and the individuals in them. It is important, therefore, that the effectiveness of organisations and the individuals in them is carefully monitored and evaluated as described earlier. This is akin to condition monitoring of plant and equipment.

Interventions to maintain organisational effectiveness can come in the form of continuous training, which is analogous to preventative maintenance, or remedial training, which is analogous to equipment repairs.

5.8.1 Knowledge management and succession planning

Nuclear power plants are designed to last for several decades and generations in the workforce. This poses two main challenges for the operating organisation. Firstly, it must have in place the means to identify the need for replacement in a timely manner. This is vitally important because of the long lead times involved in training and developing specialist personnel such as reactor operators, for example. Secondly, operating organisations must ensure that the knowledge and experience acquired by the personnel leaving the organisation is not lost with their departure. These two imperatives give rise to the need for robust succession planning and knowledge management programmes.

The IAEA has recognised that knowledge management in the nuclear industry represents an international challenge to safe operation and decommissioning of nuclear facilities. They define the challenge and their commitment to addressing it in the following way:

> 'There is clear consensus that nuclear knowledge is a strategic asset, which needs to be preserved regardless of national policies related to the utilization of nuclear power. Nuclear knowledge is needed for safe operation of nuclear facilities until they are closed down and further for their safe decommissioning and disposing of waste.
>
> Alongside other developments, the changing nuclear workforce is raising issues of "knowledge management" underlying the safe and economic use of nuclear science and technology. In recent years the nuclear workforce has been

aging, that is, more and more nuclear workers are approaching retirement age, without a corresponding influx of appropriately qualified younger personnel to replace them.

The complexity and magnitude of the problem needs a systematic approach to locate and represent the knowledge domains and to perform a critical evaluation of knowledge values.

In recognition of these and other trends, the IAEA executive bodies have called for measures to better identify the nature and scope of the problem, to understand what Member States are doing to address it, and to determine what co-operative international actions might be appropriate to enhance succession planning.

Knowledge and in particular nuclear safety knowledge is created and shared in the frame of the Agency's Nuclear Safety activities. The IAEA is pursuing a vigorous knowledge management programme to ensure that existing knowledge is fully utilized by the current generation of nuclear professionals and is effectively transferred to the work force of the future.

Focus is on knowledge generation, codification, mapping, retention and transfer. Central to the KM activities is the establishment of an environment conductive to sharing knowledge including tacit knowledge.'

Utility technical training programmes which use the SAT infrastructure devised and employed in the US are very powerful programmes for the identification of knowledge and experience required in nuclear power plants and for institutionalising it in training programmes in which it becomes sustainable.

5.8.2 Staff appraisals

Periodic staff appraisals and personnel performance management programmes are important in the process of determining training and development needs and for assessing the potential of individuals for future roles in the organisation. It is important therefore that managers and supervisors conduct them objectively and effectively as their importance merits.

5.8.3 Continuing training

Continuing training can be in two forms, refresher training or remedial training. Refresher training is systematically determined in the SAT process and is designed to ensure that personnel maintain the knowledge and skills vital to the safe and effective execution of the tasks assigned to them and to enhance their performance. Plant modifications and/or involvement in infrequently performed tasks and operations evolution can also give rise to the need for such training.

Remedial training can be determined from event investigations, peer reviews or staff appraisals, for example.

5.8.4 Standards and expectations

Personnel will, from time to time, depart from the standards expected of them and adopt behaviours that are not conducive to safe operations and high performance, through incorrect use of tools, failure to use procedures, poor housekeeping, inattention to foreign material hazards, or non-compliance with radiological protection protocols, for example. Each time a supervisor or manager observes such behaviours and standards being diminished without correcting them, they have, by their inaction, set a new and lower standard than is desirable.

A common finding in WANO peer reviews and OSART missions is a failure by management to establish and reinforce standards and expectations and a lack of presence of managers and supervisors in the field of work to provide on-going mentorship and coaching of the workforce.

Readers are referred to the WANO guideline GL2002-02, *Principles for Excellence in Human Performance*.

5.9 Basis for safe operation

The basis for safe operation is a sound design, and plant and equipment installed and operated in accordance with the design basis, all of which is subject to independent regulation.

Nuclear power plants are operated and maintained by many people on a 24-hour 365-days a year basis, so the licensee and the regulatory bodies need to establish arrangements that provide assurance that the design integrity and limits and conditions of operation are maintained at all times. Key among these requirements are:

- Rules and regulations concerning the training, qualification and licensing of personnel that ensure that only suitably qualified and experienced personnel perform the tasks assigned to them
- Technical specifications that specify the limits and conditions of plant operations and direct the operator action in normal operation and in the event of plant abnormalities
- Plant and equipment maintenance and testing schedules that ensure the plant can fulfil its design safety and performance requirements.

In addition to rules, regulations, training and experience it is vitally important that personnel conduct their work in a manner that is conducive to safe operation. In this respect it is important that the leadership of the organisation fosters a culture in which nuclear safety is accorded the significance it merits and that people act conservatively when making nuclear safety-related decisions.

5.9.1 Conduct of operations

In successful organisations the operators do much more than simply operate the plant. The operators set standards and expectations for all aspects of the plant, from performance and material condition, to housekeeping and the conduct of staff involved in its operation and maintenance. It is important therefore that operations personnel exhibit, at all times, the values, beliefs and behaviours that support high standards.

WANO Guideline 2001–02, *Guidelines for the conduct of operations at nuclear power plants*, provides further insights into the setting of standards and expectations in the conduct of operations that are conducive to high performance.

5.9.2 Communication

Operators have to conduct their work across many interfaces, between the site and external bodies such as grid control centres and emergency services, between shifts and management and support organisations across the sites. Clear, concise and effective communication is an essential attribute of operations personnel in both verbal and written forms.

WANO and INPO have developed Good Practices and Guidelines for effective communication, based on insights and experience gained in peer-to-peer exchange programmes.

5.9.3 Shift teams

There are many shift patterns employed in the conduct of operations, from four shift cycles to seven or eight. The important considerations are that they afford some continuity between shifts and provide ample opportunity for training. It is also recognised that alertness levels can vary throughout any given 24-hour period and so such factors should be taken into consideration when designing shift patterns.

In most cases shift 'teams' are kept together in the interest of promoting teamwork and effectiveness. The airline industry, however, is more concerned that familiarity between flight crews could have an adverse effect on performance, so they promote regular refreshment in their flight crews. There is considerable evidence to support the airline industry view. Shift teams that work together form group norms and habits, together with a reluctance to challenge each other's standards.

It is also true that shift teams are not exposed to the same operational experience. Normally attempts are made to cross-fertilise experience across the shifts through training scenarios, but much experience is not covered in this way and one of the few ways where it can be promoted is through regular interchange of personnel between shifts.

5.9.4 Simulator training

When managing fault situations there is no substitute for the fundamental training that operators receive to qualify them for their roles. Such training enables them to make discerning decisions based on their knowledge and experience. Simulator training (see Appendix 4) can demonstrate to the operators the relevance of such knowledge and the effectiveness of decisions they make in transient and fault situations. Much more than that, it enables the operators to practise and perfect their actions and responses to both frequently and infrequently performed plant evolutions.

In addition to improving the man–machine interfaces, simulator training is an important platform for improving human performance at both the individual and group levels. Simulator scenarios that do not cover this are missing important opportunities to enhance performance.

Regular training on simulators can be clearly demonstrated to improve both safety and reliability. Most experience suggests that simulators are most effectively utilised when they are close to the nuclear power plant sites. Although practices differ in this respect, the trend is towards site-based simulators.

5.9.5 Technical specifications

Safe operation of nuclear power installations depends on the maintenance of effective control, cooling and containment of the reactor core and its contents at all times. Designers provided systems and safeguards to satisfy these requirements. It is the role of the operator to ensure that these systems and safeguards remain available and functional through all phases of start-up, steady-state operation, transients, shutdowns and maintenance activities. The operator is also required to ensure that the plant operates within design limits and conditions that are determined in the design phase safety analysis.

Suitably qualified and experienced personnel are charged with these responsibilities; the numbers and qualifications of such people so charged are also part of the safety analysis and feature as limits and conditions to be observed in regulations.

Engineered safeguards are designed to maintain core control, cooling and containment of the reactor core. To ensure the availability and functionality of equipment, the operators conduct surveillance tests, preventative maintenance activities, inventory management and configuration management activities. These requirements feature in the form of rules and regulations for operator compliance.

From time to time, equipment will become unavailable or inoperable, operational limits will be encroached upon and transients will occur. In such circumstances it is important that the operator makes decisions in a controlled and logical manner that are consistent with the plant design limits and conditions. Technical specifications provide the operator with the basis for such decision-making.

The IAEA Safety Guide on limits and conditions (IAEA, 2000) describes the basis for the development of operating limits and conditions or technical specifications.

The US, Nuclear Regulatory Commission were the pioneers in the development and use of technical specifications. In the US, Standard Technical Specifications (STS) are published for each of the five reactor types as a NUREG-series publication. Plants are required to operate within those specifications. The regulations describe the limits and conditions to be observed in a whole range of reactor parameters.

5.10 Engineering support and design authority

Nuclear power plants are often designed and constructed by groups of companies that come together for a single or a small number of projects. The NPPs that they construct, however, will exist for several decades.

Nuclear power plants by their nature are complex. They are composed of many components and interdependent systems that must operate in a manner that meets the design intent. Over many years of operation the plant will experience many changes, equipment will become obsolete, and physical changes in the condition of materials will occur.

It is incumbent on the operating organisation, therefore, that they maintain the capability to objectively assess changes in plant condition and performance, appraise design changes and retain the knowledge base to do so. This capability will reside in those bodies of engineering personnel with the knowledge and experience to perform those duties and in the body of design data, drawings and materials acquired from architect engineers at the time of construction. Collectively this corporate intellectual feature is known as the Design Authority. The IAEA publication INSAG-19, *Maintaining the design integrity of nuclear installations throughout their operating life*, provides further detail.

5.11 References

IAEA (1991), *Safety culture*, Safety Series no. 75-INSAG-4, Vienna, IAEA.

IAEA (1999), *Safe management of the operating lifetimes of nuclear power plants*, INSAG-14, Vienna, IAEA.

IAEA (2000), *Operational limits and conditions and operating procedures for nuclear power plants*, Safety Standards series, Safety Guide NS-G-2.2, Vienna, IAEA.

IAEA (2003), *Maintaining the design integrity of nuclear installations throughout their operating life*, INSAG-19, Vienna, IAEA.

IAEA (2006a), *Fundamental Safety Principles*, Safety Standards series, Safety Fundamentals no. SF-1, Vienna, IAEA.

IAEA (2006b), The Management System for Facilities and Activities Safety Requirements. Series No. GS-R-3.

IAEA (2008), *Nuclear safety infrastructure for a national nuclear power programme supported by the IAEA Fundamental Safety Principles*, INSAG-22, Vienna, IAEA.

INPO (2004), *Principles for a strong nuclear safety paper culture*, Atlanta, GA, INPO.

US National Academy for Nuclear Training (2002), *The objectives and criteria for accreditation of training in the nuclear power industry*, ACAD 02-001, Atlanta, GA, NANT.

WANO (2001), *Guidenlines for the conduct of operations at nuclear power plants*, GL 2011-02, London, WANO.

WANO (2002), *Principles for excellence in human performance*, GL 2002-02, London, WANO.

WANO (2005), *Performance objectives and criteria*, London, WANO.

6
The need for human resources in nuclear power programmes

F. J. SÁNCHEZ, Tecnatom, Spain

Abstract: Human reliability is directly related to the competencies of the personnel. A cornerstone of a new nuclear programme is to have available, on time, enough professionals with the necessary competencies.

This chapter will help readers foresee the need for human resources, which organizations are involved, which specialities will be more demanding, the relevance of the educational system and different strategies to cope with the lack of vocations, the changing in the specialization requirements in the nuclear power plant (NPP) lifecycle, the international effort to support such challenges and some key considerations to design and implement effective initial and continuing training programmes.

Key words: human resources, competencies, knowledge management, education and training.

6.1 Introduction

Within the justification concept, safety and reliability are two cornerstone issues. In both, human reliability is always implicit. This important factor, human reliability, is directly related to the competencies of the personnel.

The International Atomic Energy Agency (IAEA, 2009a) defines 'competencies' as a 'combination of knowledge, skills and attitudes in a particular field, which, when acquired, allows a person to perform a job or task to identified standards. Competencies are developed through a combination of education, experience and training.'

No new nuclear programme will succeed if not enough personnel, having suitable competencies, are allocated on time to accomplish their duties with full responsibility. In the nuclear renaissance, solving this problem could constitute a bottle neck as important as competing for a slot in a vessel head forge or even more.

This chapter will help readers foresee the need for human resources, which organizations are involved, which specialities will be more demanding, the relevance of the educational system and different strategies to cope with the lack of vocations, the changing in the specialization requirements in the nuclear power plant (NPP) lifecycle, the international effort to support such challenges, and some key considerations in designing and implementing effective initial and continuing training programmes.

The relevance of developing strategic human resources planning at an earlier stage of the project, and the specific factors to take into consideration when planning, are addressed in Section 6.2. The core of the section reveals the appropriate staffing of the different nuclear stakeholders to carry out their mission. Among the nuclear stakeholders are included the human resources requirements of political decision makers, regulatory authorities, educational and training organizations, research centres, utilities, engineering and service companies, main suppliers and equipment vendors, construction companies, plant operators, nuclear fuel cycle and waste management companies.

Section 6.3 focuses on the nuclear education programmes, including the research and development (R&D) projects as a natural source to create new nuclear knowledge, the support provided by the national educational system, including the universities and the vocational schools, and strategies to enhance the education system to attract new vocations. Finally, national initiatives in different countries to promote nuclear knowledge are introduced.

The importance of knowledge management is discussed in Section 6.4 in connection with the changes of specialization requirements throughout the NPP lifecycle. The study is conducted in four different stages: engineering and licensing, construction and commissioning, plant operation, and decommissioning, where according to the different tasks and activities to be undertaken, the corresponding level of education and speciality for professionals, technicians and craftsmen is identified.

The benefit of international collaboration is the starting point for Section 6.5. The relevance of the Institute of Nuclear Power Operation (INPO) as a reference for personnel training and qualifications, the common education and training efforts at the level of the International Atomic Energy Agency (IAEA) or EURATOM FP-7 among others, and the international networks of excellence in education and training are the topics covered in this section.

Section 6.6 describes the main features to design effective initial and sustained training programmes based on the international standard of the Systematic Approach to Training (SAT). An important part of the section is devoted to discovering the elements to create a comprehensive training system, such as training regulatory requirements, training organization, management and staffing, training programmes and materials, instructors and training facilities and training tools, including simulators. In conclusion, the section suggests that training as a strategic tool for human performance improvement be taken into consideration.

Sources of further information, including important specialized websites, and the references used in this chapter are included in Sections 6.7 and 6.8 respectively.

6.2 Human resource requirements of the nuclear stakeholders

Nuclear technology has a specific concern which makes it different from other industries: the work in radioactive environments and, as a consequence, special requirements related to safety and radiological protection which are needed to be taken into account in all activities related to the nuclear industry. Such special requirements correspond to specific competencies which need to be thought about when planning human resources needs.

In a new nuclear programme, whether it is the first nuclear power plant or enlarging the current fleet, stakeholders should make a realistic assessment of their educational and training capabilities to develop nuclear knowledge in the quantity and quality needed.

6.2.1 The importance of human resources planning

At a very early stage (around 12 years before the first fuel loading) it is critical to plan the human resources requirements with a long-term vision. The following items must be considered within the human resources planning:

1. *Job profiles and selection criteria*. Once the decision about the project organization and the future plant organization is made, the next step will be to identify the job positions needed for the tasks to be accomplished during the different stages of the project, including plant operation and maintenance. With the detailed matrix (job position – number of candidates – year of hiring) already made, it will be useful to group the future job positions into families, such as managers, engineers, instructors or regulators, then afterwards select from them the best candidates for the different jobs.

   The number of new employees to be contracted in a specific year will depend on the capability to train them and the predicted allocation of manpower for the different tasks within the project.
2. *Training programmes*. It will be necessary to design different training programmes as the project progresses. During the first years it will be enough to provide some generic nuclear training with different content for each group to become familiar with the specific areas in the nuclear field. Table 6.1 includes typical training modules to be implemented for the different stakeholders during the first steps of a nuclear programme.

   Depending on the activity involved, more specific training will be delivered, complementing the generic nuclear part, in areas such as project management, quality assurance and quality control, licensing, norms and regulations, among others.

Table 6.1 Typical training courses during the first steps of a nuclear programme

Training content		E&P	C&C	EM	RB	O&M
NPP Fundamentals	Applied Thermodynamics and Fluid Mechanics	•	•	•	•	•
	Mechanical & Electrical Components	•	•	•	•	•
	Control & Instrumentation	•	•	•	•	•
	Strength of Materials	•	•	•	•	•
	Nuclear Physics	•	•		•	•
	Thermohydraulics	•	•	•	•	•
	Nuclear Reactor Chemistry	•			•	•
Design and Engineering	NPP Design Criteria	•	•	•	•	•
	Safety Assessment	•	•	•	•	•
	Technical Specifications	•	•	•	•	•
	Design Engineering	•	•	•	•	•
	Nuclear Safety and Safety Culture	•	•	•	•	•
Nuclear Technology	Nuclear Steam Supply System		•	•	•	•
	Reactor Auxiliaries Systems	•	•	•	•	•
	Plant Services	•	•	•	•	•
	Safeguard Systems	•	•	•	•	•
	Water-Steam Cycle	•	•	•	•	•
	Reactor Control, Limitation and Protection System	•	•	•	•	•
	Effluent treatment systems	•	•	•	•	•
Operation and Maintenance	Plant Operation. Operation Handbook	•	•		•	•
	Transient and Emergency Analysis	•	•		•	•
	Excellence in Human Performance		•	•	•	•
	Practices of Operation in Simulator	•	•		•	•
	Maintenance Management	•	•	•	•	•
	Equipment Reliability	•	•	•	•	•
	Environmental Considerations	•	•	•	•	•
RP and Regulation	Radiological Protection	•	•	•	•	•
	Nuclear Legislation and Regulation	•	•	•	•	•
	Emergency Preparedness	•	•	•	•	•
Nuclear Management	NPP Project Planning	•	•	•	•	•
	NPP Organization / Processes Management				•	•
	Industrial Safety	•	•	•	•	•
	Quality Assurance and Quality Control	•	•	•	•	•
Fuel and Decommissioning	Fuel Cycle	•	•		•	•
	Decommissioning	•	•		•	•

E&P: Engineering and Procurement. C&C: Construction and Commissioning. EM: Electrical, Mechanical and Instrumentation Equipment Manufacturers. RB: Regulatory Body. O&M: Plant Operation and Maintenance.

Some in-plant training, either tutoring or 'shadowing', is highly recommended, in order to become familiar with the allocation of systems and main equipment, the functioning of the operation departments and the internal norms and procedures. This experience can be gained in reference plants that the vendor has previously built or, if appropriate, in other plants belonging to the owner/operating organization.

3. *Task assignment within the project and professional development*. As soon as the training is finished the professionals and technicians will be assigned to a specific task within the project. The project manager should perform technical competency-based assessments, leadership development, and succession planning for future high-responsibility assignments.

According to the IAEA (2008a), the influencing factors that can reduce the human resource requirements are as follows.

- Those NPP operating organizations considering adding new nuclear units have to assess the extent to which the current workforce can be effectively utilized for the commissioning and operation of the additional units and in this way provide an opportunity to evaluate the possibility of sharing common services for the whole fleet (e.g. the Quality Assurance department or even Maintenance). It is possible to achieve as much as a 30% reduction in manpower requirements for the next reactor when maintaining an efficient organizational structure.
- Where an owner/operator owns or operates units at more than one location, a different organizational structure may be used to improve efficiency. Many functions can be centralized in the parent organization. It is common to find fleet nuclear companies that have an average of 20% fewer personnel due to the economies of scale.
- Some new advanced reactors have a more simplified design and fewer systems and components, therefore the staffing reductions for a passive light water reactor plant compared to a current nuclear plant could be about 40%.
- Finally, another factor that can affect the number of resources needed is the possibility of contracting specialized services externally. Although operating organizations tend to conduct maintenance activities themselves, rather than contracting with a vendor, there are some exceptions for outage-related work, where most operating organizations continue to rely on external support, particularly for specialized maintenance and inspections of major equipment. Engineering and technical support are other services susceptible to be contracted out. In those cases the licensee retains the primary responsibility for the safety of such operations.

6.2.2 Stakeholder staffing

Although the human resources development programme for each country has its own unique characteristics that should be identified considering the above-mentioned factors, in the following paragraphs the human resources needs for different stakeholders will be analysed according to their main mission.

The ranges presented in the following paragraphs should be interpreted as indications of orders of magnitude of the number of specialists required for each group of activities for a new NPP with a single unit. Most data have been extracted and adapted from IAEA (2007a).

Table 6.2 summarizes the human resources requirements according to different functions or activities to be accomplished during the implementation of the nuclear programme. Statistics regarding future nuclear employment in the USA can be found in Clean and Safe Energy Casenergy Coalition (2009).

Political decision makers

The long-term commitment and involvement of the corresponding governmental organizations (ministries such as the Department of Trade and Industry, Energy Planning Commission, etc.) is very important in order to guarantee the adequate development of the nuclear infrastructure needed for the country.

This commitment will provide credibility for the nuclear programme investors whose involvement is critical to support policies on human resources development, technology selection ('justification process'), licensing and regulation development, national infrastructure needs and the necessary international agreements related to nuclear power. A minimum number of highly qualified personnel will be necessary at this level.

Regulatory authorities

The nuclear regulatory body needs to be created or expanded, with responsibility for defining all safety, safeguards and security requirements according to codes and standards and ensuring that they are met. The regulator will have a relevant role during the licensing process of a new NPP.

Assistance in developing human resources may be provided by the regulatory body in the country of origin of the supplier or other regulatory bodies, and complemented by the IAEA and other international organizations.

Table 6.2 Human resource requirements of the nuclear stakeholders

Stakeholder	Activities	Staff[1]	Observations
Political decision makers	• Power system planning • Regulatory framework • National infrastructure	15–25	
Regulatory authorities	• Codes and Standards • Licensing • Inspection	45–65	Licensing effort: 150–200 man-years
Utility	• Site selection • Human resources plan • Bid specification and evaluation	50–100	The needs start to increase strongly when the commitments are made (letter of intent, contract, etc.)
Training organization	• Training system organization, materials development and delivery of training sessions • Full-scope simulator	25–45 15–25	Qualified instructors plus simulator engineers
Engineering and services companies	• Conceptual design • Basic design • Detailed design • Equipment and plant specifications and drawing • Physical protection • Procurement	250–350 10–12 20–50 25–40	Project engineering work requires some three million man-hours of effort over a relatively short period (3–5 years)

Table 6.2 Continued

Stakeholder	Activities	Staff[1]	Observations
Main suppliers and equipment vendors	• Equipment and component manufacturing • Equipment erection • Piping supporting and welding • Non-destructive testing	3000	Manufacturing and construction are the activities that have by far the largest manpower requirements, of the order of 6000 people during its peak period
Construction companies	• Construction, erection and installation of plant buildings • Component and systems erection and installation	2000 1300	
Plant operator	• Commissioning • Operation and maintenance	40–50 400–1000	Major support during commissioning by the equipment manufacturers
Nuclear fuel cycle company and waste management	• Exploration • Fuel fabrication • Waste management	100–200	
Decommissioning organization	• Decontamination • Dismantling • Asset recovery • Waste processing, storage and disposal	500–1000	

[1] The ranges presented in the table should be interpreted as indications of orders of magnitude of the number of specialists required for each group of activities for a new NPP with a single unit.

Educational and training organizations

The educational system (universities and vocational schools) will support the baseline of highly qualified professionals in the different specialities required by the nuclear programme. The nuclear training organizations (nuclear institutes and training centres) will enhance the specific nuclear competencies of the graduates from the educational system, giving them a more practical approach to their future duties.

The training organization should be established as soon as possible, since it is the pipeline to generate nuclear knowledge for the new staff. Training instructors should therefore be one of the first groups of people to be hired in the human resources planning. Senior instructors from existing training centres or external support (from the main supplier or specialized companies) will be required to train new instructors.

The availability of a plant-referenced simulator well in advance of fuel loading not only provides a unique tool for training control room personnel, but also is very useful for the development and validation of operating procedures, commissioning tests, and training of other plant personnel, as well as for a variety of activities including engineering, design modifications, configuration management and licensing. Appendix 4 discusses training in these plant-referenced simulators.

Senior instructors can cooperate in the specification of the simulator and the writing of the acceptance test procedures, but additional hardware and software engineering will be needed to develop the simulator model and panels.

Research centres

Nuclear research centres will give the scientific and technical support to nuclear development as well as promoting the research and development for current and foreseen future problem-solving and technological innovation. Additionally these organizations will facilitate the transfer of technology. Chapter 7 describes the needs and roles of such nuclear research centres.

The participation in international research and development projects, excellence networks and specialized forums, such as those identified in Section 6.5, is another important source of nuclear knowledge and international relationships.

Utilities

Utilities will have a crucial role to play in the development of the nuclear programme: site and technology selection, economic feasibility and financing, project planning, licensing, construction, commissioning and operation of the future nuclear facilities.

During the preparation phase the overall manpower needs are relatively modest, mostly orientated towards directing, coordinating and registering data, but do involve a large number of organizations (including political decision makers). Although the manpower required is relatively low, they need to be highly qualified professionals. The relevant staff should preferably have professional experience in the coordination and performance of complex interdisciplinary studies. The needs start to increase strongly when the commitments are made (letter of intent and contract) to build the plant. The involvement of a knowledgeable consultant is recommended.

Engineering and services companies

Project engineering work requires a huge effort over a relatively short period of time. This can be done either by the utility itself or by its architect-engineer or main contractor or by a combination of efforts from some of the participants. In any case, the minimum involvement of the utility (review and approval) will amount to about 50,000 engineering man-hours.

The conceptual design task can involve from 20 to 30 experienced engineers and technicians for a period of up to $2\frac{1}{2}$ years. It is a task that should normally be completed about seven years before commercial operation of the plant. For the independent review, a total effort of at least 2000 man-hours would normally be required.

The next task of design engineering can be divided into two: basic and detailed design. Basic design engineering can involve 300,000 to 500,000 man-hours for a period of 6–12 months. The task of detailed design engineering involves about 2,500,000 man-hours of effort during a period of some 3–5 years. For specifications work there should be at least 10–12 engineers.

Adequate physical protection of the plant and nuclear material requires mainly administrative and security functions.

Procurement could also be handled directly by project management or by project engineering. A minimum number of professionals and technicians would be required for a centralized independent procurement unit.

Main suppliers and equipment vendors

Main suppliers and equipment vendors that own the technology play a relevant role in the first NPP project in transferring the technology know-how.

Equipment manufacturing is the largest block of man-hours in a nuclear project, approximately 20 million man-hours for a 1000 MWe plant. The overall manpower requirements for the manufacture of equipment and components are estimated to be of the order of 3000 professionals, technicians and craftsmen.

Construction companies

Depending on the difficulties of the particular site, the tasks to be performed for site preparation will require normally 50 to 150 craftsmen and labourers during this stage as well as 10 to 20 professionals and managers, who have previously performed similar duties. The number of craftsmen and labourers could increase by as much as a factor of five for exceptionally difficult sites, taking into account that the above tasks should be completed as quickly as possible.

Usually the peak of concrete work occurs during the first year of construction and the peak for interior finish and masonry work is in the third year. The overall manpower requirements for civil construction will generally peak during the second and third year of construction after which they will gradually decline.

To coordinate, manage and expedite component installation requires an experienced team at the site of at least 25 professionals during the peak period.

For equipment, component and systems erection and installation, a peak workforce (about four years after construction starts or earlier with the advanced reactor construction) of the order of 1300 people would be required. Many of the welders must be qualified for specialized cover-gas equipment. For difficult sites (climate, high rate of personnel turnover, low worker efficiency) the overall quantity of manpower could be 20–50% higher, or it can easily double.

Plant operators

The major part of NPP commissioning covers a period of about 2–3 years from the finished erection of the first systems (electrical energization) up to the start of the commercial operation of the station. About 40–50 professionals are usually assigned to perform the tasks for this activity. Major support will be provided by manufacturers with the participation of plant operation and maintenance personnel.

The overall manpower requirements for the stage of plant operation are not so much directed by the plant output capacity as by the policies regarding the uses of external contractors. For guidance, the manpower requirements can be defined as an average of one worker for 1 megawatt electrical of gross capacity of the plant (680 MWe gross capacity would require approximately 680 workers). This average value of workers per MWe is not linear and tends to be as low as 0.7 as the plant gross capacity increases, especially for multiunit stations.

Nuclear fuel cycle and waste management companies

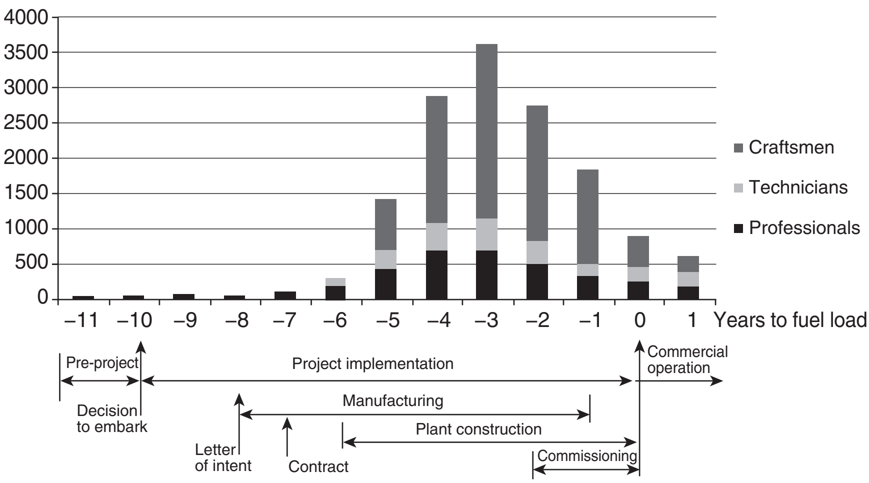
The minimum essential activities which must be performed by a country itself are:

- Procurement of uranium, uranium conversion and enrichment and fuel fabrication, involving 4–6 persons.
- Fuel management at the power plant and disposal of spent fuel. It is usually the responsibility of the owner of the nuclear power plant to carry out these tasks.
- Waste management: without reprocessing, the back-end activities will possibly require 100–200 people.

If additional fuel cycle activities are taken up in the country, such as uranium exploration and production, or fuel element fabrication, specific organizations and the corresponding manpower will be required to carry out the tasks.

At the end of the plant life and for decommissioning purposes, including decontamination, dismantling, asset recovery, waste processing, storage and disposal, around 500–1000 staff will be necessary.

An example of the overall manpower requirements during the different stages of a nuclear power project is illustrated in Fig. 6.1 adapted from



6.1 Manpower loading for a nuclear power project (example: average case based on 1000 MWe PWR plants under construction)

IAEA (1980). The data do not include resources for equipment and component manufacturing.

During the pre-project and early implementation phases, a relatively small number of highly qualified professionals are needed. The requirements start to substantially increase when commitments are made (letter of intent, contract) to install the plant. The activities which have by far the largest manpower requirements are manufacturing and construction.

6.3 High-level nuclear education programmes

According to the remarks made by the Honourable Peter B. Lyons, Commissioner from the US Nuclear Regulatory Commission at the 2008 International Congress on Advances in Nuclear Power (ICAPP'08), 'Creating, sustaining, and growing a population of educated, trained, and experienced personnel from which the nuclear industry need to recruit in order to accomplish their goals is a challenge among government, industry, and academia'.

It is widely recognized that national development in the nuclear industry requires a scientific and technological infrastructure. Such an infrastructure is mainly found in:

- National and private research and development (R&D) institutes
- Institutes and laboratories for standardization and calibration
- Higher education institutions
- Vocational schools for practitioners and professional training centres
- Scientific academies and professional associations
- National industry.

All these organizations create knowledge in one way or another but three of them are particularly important: R&D institutes, higher educational institutions and vocational schools.

6.3.1 National and private research and development (R&D) institutes

The development of a national academic programme for the education of the necessary scientists, engineers and other technicians to support technical research would also be expected to be in place as part of the commitment to the development of the required national capabilities.

To build new nuclear knowledge it is particularly interesting to participate in R&D projects related to nuclear disciplines, such as nuclear fuel, nuclear materials, management of radioactive waste, nuclear safety or radiation protection.

Examples of international R&D platforms are:

- The IAEA International Project on Innovative Nuclear Reactors and Fuel Cycles (INPRO), established in 2001
- The European Commission within the Seventh Framework Programme (2007–2013) for nuclear research and training activities
- The European Union Sustainable Nuclear Energy Technological Platform (SNTP)
- The 'Global Nuclear Energy Partnership' (GNEP), founded in September 2007
- Generation IV International Forum (GIF), since January 2000.

6.3.2 Higher education institutions

For a country with nuclear interests the development of a national nuclear education programme involving government agencies, laboratories and research facilities, helping to attract and develop initial experience and knowledge in nuclear technology, is highly recommended.

Nevertheless, and while the national academy system is mature enough and some nuclear programmes have been developed at the university level, it is possible to rely on other countries to offer and maintain these types of educational programmes.

What is, without any doubt, necessary from the beginning is to have a good general engineering (electrical, mechanical, control and process) and physics education infrastructure, producing high-quality graduates, who can then be trained in appropriate nuclear subjects, either within the industry, in cooperation with other training or academic providers, or even as part of the turnkey contract by the vendor.

6.3.3 Vocational schools for practitioners and professional training centres

A new nuclear programme needs thousands of skilled craftsmen, in addition to engineers, such as welders, boilermakers, iron workers, pipefitters, construction labourers, millwrights, electricians, carpenters, insulators and heavy equipment operators. Hence the importance of vocational schools and apprenticeship programmes.

These specialists require specific training in quality assurance, safety and radiological protection, if they are to work in the nuclear field. The worker is required to achieve nationally accepted standards of competence in order to satisfy the vocational training requirement. The worker's competence is assessed while observing his or her performance in various standard tasks, by assessing knowledge and understanding (typically by using oral and written questions), and by collecting other evidence about the worker's competence.

Another approach is through apprenticeship programmes for young students leaving secondary education who gain, during a certain period of time, real work experience and some complementary nuclear training. The emphasis throughout this training period is on exposing the apprentice to as many of the different facets of nuclear work as possible, while ensuring that the learning process is fully supervised with respect to safety. While set standards are required to be met in order for the apprentice to successfully achieve a formal apprenticeship qualification, this can also be considered as human resources development, using this qualification as the first step in the process of building a successful career in the industry.

6.3.4 Strategies to enhance the education system

The IAEA (2009a) suggests that the existing national educational institutions can enhance the support they provide for the development of human resources for the nuclear industry in different ways, such as by:

- Developing new, or realigning existing, nuclear engineering and science-related degree curricula jointly with nuclear responsible organizations to ensure alignment with future needs.
- Establishing working 'councils' with academic, government and industry representation to oversee the development of nuclear sciences training and development programmes nationally.
- Developing partnerships with appropriate programmes in countries with mature nuclear power programmes, and then using these relationships to develop new programmes or gain accreditation of existing programmes.
- Developing 'fellowship' programmes, whereby national undergraduates get the opportunity to pursue a portion of their study in a country with a well-developed nuclear power programme.
- Providing 'work placement' opportunities whereby students can work in the various organizations (operating organization, regulatory body or support organizations) for a period from a few weeks up to a year, to gain insight and experience in the organizations.
- Providing support funding for an appropriate 'chair' or Head of Faculty position (e.g. engineering, physics, nuclear sciences) at one of the better engineering universities.
- Funding relevant research, such as material studies, fatigue mechanisms or diagnostic techniques, among others; this will be of real benefit to the nuclear industry while at the same time attracting and encouraging high-quality academic staff.

Additional recommendations selected from the Nuclear Energy Agency, NEA (2000) are:

- Create a pre-interest in the nuclear domain: include steps such as advertisements aimed at undergraduate candidates; high school 'open days' at campuses or research facilities; regular reactor visits for students; newsletters, posters and web pages; summer programmes; preparation of a resource manual on nuclear energy for teachers; recruiting trips and nuclear introduction courses for freshmen; and conferences given by industry and research institutes.
- Add content to courses and activities in general engineering studies: increase emphasis on nuclear physics and applied physics courses; organize seminars on nuclear in parallel with the existing curriculum using speakers external to the university; discuss employment potential and professional activities and call attention to the environmental benefits of nuclear power.
- Increase pre-professional contacts. Encourage the participation of students in the activities of the local nuclear society and its 'young generation' network.
- Provide opportunities for high school students and undergraduates to work with faculty and other senior individuals in research situations.

Finally, a very useful strategy is to organize postgraduate nuclear courses (university specialization courses or Masters) in cooperation with the university and the nuclear industry. This can get the best of both worlds: theory and academic rigour at the high scientific level from the university teachers, and the practical application to the industry from the company experts.

If there are not enough students and/or teachers in the country to organize an independent training programme, cooperation between different countries is encouraged, utilizing available international assistance, e.g. Appendix 3 on IAEA programmes gives information on IAEA training materials. The World Nuclear University or Regional Networks, described in Section 6.5, should also be considered.

6.3.5 Specialities needed in the nuclear renaissance

In order to analyse the technical manpower required in the development of a nuclear programme, it is convenient to divide it into three primary categories: professionals, technicians and craftsmen.

Professionals

This refers to all managerial and technical personnel whose normal minimum formal educational requirement is a Bachelor of Science (B.Sc.)

degree or equivalent from a recognized or accredited institution of higher learning (i.e. university or college).

Professionals are obviously the primary component of the manpower required to plan and supervise the implementation of, and assume responsibility for, all activities within the nuclear power programme. They also require the longest lead times for their development.

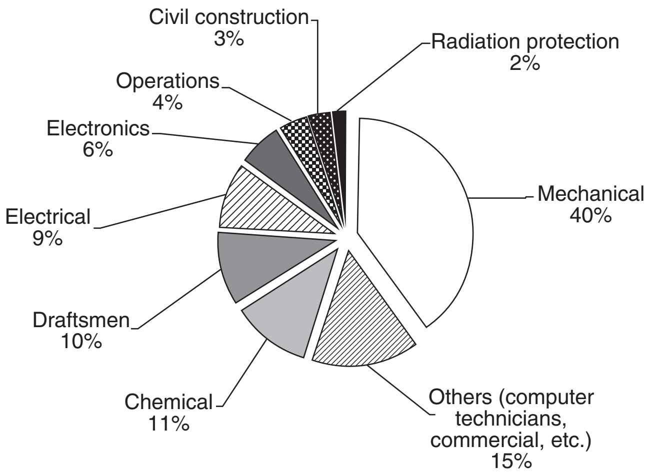
Many activities that involve professionals need a relatively high number of mechanical and chemical engineers. This is to be expected in a technology involving power plants with large high-technology equipment requirements and a fuel cycle with complex chemical processes. The level of educational requirements of professional manpower for the main activities involved in a nuclear power programme requires different specialities, such as:

- Master of Science (M.Sc.) / B.Sc. in engineering with the following:
  - Nuclear engineering
  - Power plant engineering
  - Mechanical engineering
  - Electrical engineering
  - Electronics engineering
  - Chemical engineering
  - Civil engineering
- M.Sc. / B.Sc. in metallurgy, physics and chemistry
- B.Sc. in geology, hydrology, meteorology, ecology, biology and seismology and environmental sciences
- Computer programming technician
- Bachelor of Arts (B.A.) in economics and business administration
- Master of Arts (M.A.) in law
- M.A. in commerce
- Accountants
- B.A. in journalism.

Technicians

This refers to all sub-professional level personnel who have scientific and technical training at an appreciable level beyond the 12th grade but less than the minimum educational requirement of the professional level. Technicians are trained persons who are broadly knowledgeable in such disciplines as mechanical, chemical, electrical or electronic technology, or who have specialized knowledge and capability in specific fields such as radiation protection, instrumentation, materials testing, quality control and process control.

A typical distribution of the technician-level workforce for a nuclear power programme might be approximately as shown in Fig. 6.2.



6.2 Technicians in a typical nuclear power programme

Craftsmen

This refers to those skilled workers who, by a combination of training and experience (usually through an apprenticeship), are well qualified to perform specific types of tasks, operate specific classes of equipment or perform specific operations.

Craftsmen are mainly required for plant construction and for the manufacture of equipment and components. It should be noted that qualified pipe fitters and welders each represent approximately 15–20% of the total craftsmen workforce during the construction stages.

Examples of this type of worker would include boilermakers, carpenters, concrete workers, electricians, insulators, iron workers, millwrights, operators of heavy equipment, painters, pipe fitters, sheet-metal workers and welders.

6.3.6 National initiatives in different countries

Some countries are carrying out national education initiatives to promote nuclear knowledge, such as the examples in the USA, the UK, Japan and France described below.

USA: NEUP (2009)

The US Department of Energy's Office of Nuclear Energy (DOE) created Nuclear Energy University Programs (NEUP) in 2009 to consolidate its university support under one programme. NEUP funds nuclear energy research and equipment upgrades at US colleges and universities, and provides scholarships and fellowships to students. DOE personnel in Washington DC oversee the programme and the Idaho-based NEUP Integration Office administers the awards. NEUP's goals and objectives are to support outstanding, cutting-edge and innovative research at US universities by:

- Attracting the brightest students to the nuclear profession and supporting the nation's intellectual capital in nuclear engineering and relevant nuclear science, such as health physics, radiochemistry and applied nuclear physics
- Integrating research and development (R&D) at universities, national laboratories and industry to revitalize nuclear education
- Improving university and college infrastructures for conducting R&D and educating students
- Facilitating the transfer of knowledge from the aging nuclear workforce to the next generation of workers.

UK: NTEC (2008)

The Nuclear Technology Education Consortium (NTEC) is a consortium of 12 UK universities and other institutions providing postgraduate education in nuclear science and technology.

The structure and content of the programme, which leads to qualifications up to Master's level in nuclear science and technology, was established following extensive consultations with the UK nuclear sector, including industry, regulators and government departments among others.

All training modules are delivered by direct teaching but some have been converted into a distance learning format as an alternative method of delivery to provide greater choice for students. The first modules in this format were launched in September 2008.

Modules are generally delivered on the campus of the providing institution. Students seeking a postgraduate qualification register with the university of their choice and visit other members of the consortium to attend their selected modules. The programme is coordinated by the Dalton Nuclear Institute at the University of Manchester.

Japan: GoNERI (2007)

The University of Tokyo Global COE Program: Nuclear Education and Research Initiative (GoNERI) carries out research and education activities in three areas: nuclear energy, radiation applications and the social aspects of nuclear engineering.

Among GoNERI's activities are the advanced summer school in radiation detection and measurements in cooperation with the University of

Berkeley (USA), the international summer school of nuclear power plants and young generation workshop, and the study on the international nuclear fuel cycle framework from a nuclear non-proliferation viewpoint.

France: CFEN (2008)

In 2008, the French Minister for Higher Education and Research created a co-ordination committee for nuclear education and training in order to ensure the expansion of the French nuclear energy sector through the renewal of its workforce. This committee, recently renamed the '*F*rench *C*ouncil for *E*ducation and Training in *N*uclear Energy' (CFEN), assesses the adequacy between the education offer, the student population in different curricula and the industrial/research needs, advises the Office of Higher Education on opening new academic curricula, informs students of various educational curricula and possible professional careers and opportunities in nuclear power technology, coordinates the international recruitment of students, and promotes international curricula such as the new International Master of Science in Nuclear Energy starting in Paris in 2009.

The members of the CFEN include representatives of government authorities in education, research and industry, of academic institutions (universities and engineering schools), of the chief industrial actors (AREVA, EDF, GDF-SUEZ, ANDRA, and subcontractors), and of the main nuclear R&D public institutions: CEA and IRSN.

More than 20 chief universities and engineering schools, distributed all over the country though with many located in the Paris area, provide nuclear engineering-related education programmes. CEA/INSTN (Institut National des Sciences et Techniques Nucléaires) also plays an important role in this field through its establishments located in Saclay, Cherbourg and Cadarache.

6.4 Changing specialization requirements in the nuclear power plant lifecycle

A viable nuclear knowledge culture needs constant attention throughout the different stages of the nuclear lifecycle, which implies nuclear knowledge management.

Adequate numbers of competent and motivated personnel must be available during any phase of a nuclear programme. From the regulatory perspective, the licensing requirements define that the licensee needs to be able to demonstrate that adequate numbers of competent personnel are available, until the facility is finally removed from regulatory control.

The knowledge and skills necessary to purchase, construct, license, operate, maintain and comply with regulations of a nuclear power plant are spread across most scientific and engineering disciplines. Specific considerations for the nuclear industry include:

- Additional knowledge and appreciation of the increased attention to detail in order to ensure operational safety, security and radiation protection are vital and require a heightened attention to the quality of major systems and equipment.
- Expertise in nuclear physics and nuclear materials science for reactor operation and fuel cycle management.
- Finally, along with the technical skills, there must be a strong commitment to safety culture, which instils personal responsibility for the safety of all individuals involved in the programme.

6.4.1 The importance of knowledge management: key considerations

Lifespans of nuclear power plants significantly exceed the working life of a single generation of plant staff. This presents the challenge of retaining knowledge and operating experience as the workforce ages and new generations of personnel are hired. This area, termed 'knowledge management', is particularly important in establishing both information databases and the transfer of knowledge and experience to new personnel.

Key considerations related to knowledge management are:

- Selection, copying, and reclassification of documents from one step applicable to the next, such as licensing documents, construction details, operating experience and design modifications
- Retention of personnel with knowledge of especially important aspects applicable to the next step. Technicians involved in commissioning for the operation and maintenance of the plant, or operators or key maintenance technicians for decommissioning
- Programme for the transfer of know-how to other workers, through on-the-job training, mentoring and other techniques to complement the formal training.

The subject of knowledge management is treated in more detail in IAEA (2004).

6.4.2 Specialization requirements in different stages of the NPP lifecycle

Analysis of the specialization requirements will be divided into different phases according to the important activities to be accomplished in the nuclear programme from the preparation phase until the commissioning and the initial commercial operation.

Engineering and procurement

To be prepared to issue a bid request for the first nuclear power plant, the staff need to be in place with a basic knowledge of the specific technologies chosen to prepare the bid specification and to establish the evaluation criteria. Staff should be available to evaluate and select a winning candidate from a technical, management, business and economic perspective.

Although operators and maintenance technicians do not have to be in place for the moment, some knowledge of operational and maintenance requirements needs to exist within the team. Initial education and training for the remaining resources to fully support plant operation should begin at this time.

The IAEA (2007b) identify the specific human resource needs at this stage including:

- Business and technical expertise for site qualification and preparation of the construction permit request
- Political and social expertise for public communication
- Technical and regulatory expertise to develop and implement regulations, codes and standards for plant licensing, site approval, operator licensing, radiation protection, safeguards, physical protection, emergency planning, waste management and decommissioning
- Business and technical expertise for fuel cycle procurement and management
- Expertise to conduct training programmes for construction and project management
- Plans to fully staff and train the regulatory body for operational oversight
- Plans to fully staff and train operating, maintenance and support organizations
- Plans to develop future expertise in all relevant areas, including any needed enhancements to the national educational institutions.

Professionals during design periods are needed primarily for project management and engineering. In addition, manpower is required to perform the supporting activities: NPP project planning and coordination, regulatory and licensing activities and fuel cycle activities, among others.

The conceptual design task will involve experienced engineers and technicians. At the end of the conceptual design task all major characteristics of the plant should be defined. The results take the form of systems descriptions, conceptual drawings, data compilation and preliminary licensing information. These results should be subjected to an independent review by experienced engineers who are senior professionals not previously involved in the conceptual design development. Consultants who have previous experience on other similar projects may also be utilized.

For basic and detailed design a high level of engineering practice is required. The preparation and review of equipment and component specifications constitute an important part of the detailed engineering task. The result of the design work will ultimately be passed on to sub-contractors in the form of equipment and plant specifications and drawings. The production of these documents is a major effort involving not only the design engineers but also other technical personnel knowledgeable in the areas of manufacturing, materials, engineering, licensing and quality assurance. For specifications work, in particular, there should be engineers with prior experience of writing specifications to lead the task of specifications development.

Table 6.3 summarizes the specialization requirements during the engineering and procurement stages.

Construction and commissioning

Site preparation will require craftsmen and labourers, as well as professionals and managers, who have previously performed similar duties. Most of the staff during plant construction (about 85%) will be technicians and craftsmen. In the nuclear power industry, the requirements for unskilled labour are very low (of the order of 10%) although in some countries their proportion may be considerably higher, mainly owing to local labour practices and employment policies. The construction, erection and installation of plant buildings will require one or more qualified civil engineering and construction firms with skilled and experienced workers.

For the manufacture of equipment and components there will be needed mechanical and electrical technicians, foremen and craftsmen, labour and administration.

To coordinate, manage and expedite component installation requires an experienced team. For equipment, component and systems erection and installation most of the required workforce will be technicians and craftsmen. Many of the welders must be qualified for specialized cover-gas equipment. At least 30% of the mechanical technicians and 10% of the electricians should have knowledge and familiarity with relevant codes, standards and criteria.

Core components erection is of a special nature and requires precision tolerances and aligning to close accuracies. Qualification of procedures by mock-ups and qualification of personnel are important. This stage of the construction provides the best possible opportunity to complement the

Table 6.3 Specialization requirements during engineering and procurement

Tasks and activities during the different stages of the NPP lifecycle	Requirements of education
Pre-project activities:	
• Power system planning	B.Sc. in engineering, preferably electric power; economics (B.A.) and computer programming technicians
• Feasibility studies	B.Sc. in engineering; Economics and Law degrees
• Site survey and qualification	B.Sc. or B.A. in engineering, geology, hydrology, meteorology, ecology, biology and seismology
• Project management, supervision, quality assurance, safety and licensing	M.Sc. or B.Sc. in engineering (nuclear, mechanical, electrical and electronics), metallurgist, physicist and chemistry, draftsmen
• Administration and public relations	B.A. in economics and business administration, accountants and B.A. in journalism
Project engineering:	
• Project engineering management and supervision	M.Sc. or B.Sc. in engineering (nuclear, mechanical, electrical, electronics and chemical)
• Nuclear engineering	M.Sc. or B.Sc. in engineering (nuclear, mechanical or chemical), metallurgist, physicist
• Civil engineering	B.Sc. in civil engineering
• Mechanical engineering	B.Sc. in mechanical engineering, draftsmen and mechanical design
• Electrical engineering	B.Sc. in electrical engineering, draftsmen and electricians
• Instrumentation and control engineering	B.Sc. in engineering (electronics, electrical or computer)
Procurement:	
• Procurement management	M.Sc. in engineering or M.A. in commerce, law or business administration
• Markets and coordination	M.Sc. in engineering (mechanical, electrical or nuclear), B.Sc. in engineering or B.A. in commerce
• Bidding and contracting	M.Sc. in engineering or M.A. in commerce, law and commercial technicians
• Monitoring and expediting	B.Sc. in engineering or B.A. in commerce
Quality assurance and quality control	M.Sc. and B.Sc. in engineering and technicians (mechanical, electrical and electronics)

Adapted with permission from IAEA (1980), Table 1.12-1 to Table 1.12-10 Manpower Requirements and Technical Qualifications, on pp. 133–184 of the Technical Reports Series No. 200, *Manpower Development for Nuclear Power: A Guidebook*, IAEA, Vienna.

training of the future plant maintenance personnel, who should actively participate in the erection and installation effort and would thus gain further experience. In addition, the contractors and subcontractors and their skilled personnel would provide a very valuable manpower source for future plant maintenance and, in particular, for major overhauls, repairs or modifications.

During commissioning the specific human resource requirements according to the IAEA (2007b) include:

- A fully staffed nuclear power plant operation, maintenance and technical support organization
- A fully staffed regulatory body with specific expertise in operating plant oversight
- Succession and personnel development planning to sustain the competence of all areas of the national nuclear programme
- Enhanced educational opportunities for nuclear science and technology
- Enhanced training programmes for the development of operators and technicians.

Major support during commissioning is to be provided by engineers and technicians from the equipment manufacturers. In addition, the plant operation and maintenance personnel participate actively in the commissioning of the plant; such participation is in fact considered to be the last essential part of their training. It is necessary to emphasize that during commissioning the responsibility will be transferred from the construction team to the operating organization.

Plant operation

The operating organization responsible for an NPP has a staff that collectively has a variety of scientific, engineering and other technical backgrounds in fields needed to effectively and safely operate and maintain the plant. These include nuclear engineering, instrumentation and control, electrical engineering, mechanical engineering, radiation protection, chemistry, emergency preparedness, and safety analysis and assessment. There is a need to have access to national or international expertise to support the NPP operating organization and regulatory body in scientific areas such as neutronics, physics and thermohydraulics and in technical areas such as radiation protection, radioactive waste management, quality management, maintenance and spare parts management.

In addition to the required scientific, engineering and other technical education, normally the relevant staff need three or more years of specialized training and experience prior to the initial fuel loading of an NPP. For implementation of a first NPP project, much of this specialized training and experience can be included as part of the contract with the supplier of the NPP technology. It is necessary for the operating organization to establish the rigour, culture, ethics and discipline needed to effectively manage nuclear power technology with due regard to the associated safety, security and non-proliferation considerations (IAEA, 2007c).

Table 6.4 includes the specialization requirements during the stages of construction, commissioning and plant operation.

Decommissioning

The activities undertaken during decommissioning, following any routine programmes of defuelling or facility system flushing, generally comprise a formal sequence of non-routine tasks. To ensure that these tasks are completed with respect to safety, programme, quality and cost considerations, it is important to identify the change of emphasis in the training requirements as the transition from operations to decommissioning occurs.

The risk of losing knowledge, both explicit and tacit, increases with the time passed. The problem is compounded by the fact that efforts to identify the information requirements for decommissioning are not usually an organized and consolidated activity and may not be appreciated by organizations operating the nuclear facility. For these reasons, it is important to

Table 6.4 Specialization requirements during construction, commissioning and plant operation

Tasks and activities during the different stages of the NPP lifecycle	Requirements of education
Construction:	
• Plant construction management	M.Sc. in engineering (civil or mechanical)
• Plant construction supervision	B.Sc. in engineering (mechanical, electrical, electronics, civil)
• Commercial and administration supervision	B.Sc./B.A. in business administration, accounting
• Construction, erection, installation of buildings, structures, equipment and components	B.Sc. in engineering (mechanical, electrical, electronics, civil) and technicians (mechanical, electrical, instrumentation, civil construction, accountants, draftsmen, computer) and craftsmen (boilermakers, carpenters, concrete workers, electricians, insulators, iron workers, millwrights, operators of heavy equipment, painters, pipe fitters, sheet-metal workers, welders)

Table 6.4 Continued

Tasks and activities during the different stages of the NPP lifecycle	Requirements of education
Commissioning:	
• Commissioning management	M.Sc. in engineering, preferably mechanical
• Commissioning supervision	B.Sc. in engineering (mechanical, electrical, nuclear, chemical)
• Commissioning tasks such as development of procedures, performance of tests, preparation of reports; adjustments, modifications	B.Sc. in engineering (mainly mechanical, electrical, nuclear, also electronics, chemical, civil); physicist; chemist; metallurgist and technicians and craftsmen in specific field of activities
Plant operation and maintenance:	
• Plant, operation, safety and training management	M.Sc. in engineering
• Shift supervision	B.Sc. in engineering, preferably electrical or mechanical
• Control room operation	Technicians (might be B.Sc. in engineering), electrical or mechanical
• Field operation	Technicians (electrical, mechanical)
• Maintenance management	B.Sc. in engineering (preferably mechanical)
• Maintenance engineering	B.Sc. in engineering
• Performance of maintenance	Mechanical, electrical and instrumentation and control technicians and mechanical crafts, electricians, electronics and civil crafts
• Nuclear safety engineering	M.Sc. in engineering
• Industrial safety engineering	B.Sc. in engineering
• Radiation protection management	M.Sc. in engineering or physicist
• Radiation protection monitoring	Technicians
• Training	B.Sc. in engineering, physicists, chemist and technicians (mechanical, electrical, radiological protection)
• Technical supporting services	B.Sc. or M.Sc. in engineering (nuclear, mechanical, electrical, electronics, chemical); physicists, chemists and technicians (mechanical, electrical, electronics, chemical, computer, draftsmen)
• Quality assurance	B.Sc. in engineering (preferably mechanical) and technicians (mechanical, electrical, civil, welding)

Adapted with permission from IAEA (1980), Table 1.12-1 to Table 1.12-10 Manpower Requirements and Technical Qualifications, on pp. 133–184 of the Technical Reports Series No. 200, *Manpower Development for Nuclear Power: A Guidebook*, IAEA, Vienna.

consider decommissioning as a phase in the lifecycle of a nuclear facility and to preserve during operation the records and information that might be useful after shutdown.

Training has an important role during the transition to decommissioning, when the detailed design of the decommissioning project and its organization are being developed. Training can be an effective tool to transmit information stored during the operation of the facility to the decommissioning organization and its personnel. In the same way, training is also essential during the planning and performance of specific decommissioning tasks, particularly during the detailed planning of each work package, which usually relies on a sound knowledge of the configuration and the operational history of the systems to be dismantled. Thus, in this phase, the training of work supervisors, health physics personnel, ALARA technicians and industrial safety personnel, and other personnel, can be accomplished.

According to IAEA (2008b), typical subject matter for generic safety and other training is as follows:

- General employee training in radiation safety, industrial safety, fire safety and emergency planning
- Radiation worker safety training
- Respirator (full-face, half-face, self-contained breathing apparatus) training
- Airline suit training
- Electrical safety training
- Confined space training
- Crane, hoisting, and rigging training
- Lockout and tagout training (safe system of work)
- Fire watch training
- Forklift safety training
- Human performance awareness fundamentals training
- Peer and self-checking
- Project reviews and pre-job briefings
- Use of power tools
- Manual handling
- Basic first aid
- Working at height
- Chemical/hazardous material handling.

Finally, contractors are used more in the 'worker' group to provide specialist support and to satisfy peak labour demands. The training for contractors is no less onerous than that for the client organization workers, and in many cases may be greater due to the non-familiarity of the contractor worker with the working environment.

In accordance with the tasks to be performed during decommissioning, different positions could be found such as operators, technicians (radiological protection technicians and chemistry technicians), maintenance personnel (electrical, mechanical and instrumentation and control technicians), craft personnel (welders, pipefitters, carpenters) and supervisors of the above categories, with an academic profile similar to those described in the preceding paragraphs.

6.5 International experience

At the beginning of the twenty-first century, when the nuclear generational change started, many international organizations declared their concerns regarding the lack of enough candidates to substitute for the retiring workforce, due to declining interest among students in nuclear matters. This situation would threaten the preservation of nuclear knowledge in the world.

In the report *Nuclear education and training: Cause for concern?* (NEA, 2000) the Nuclear Energy Agency alerted the national authorities responsible for education and nuclear safety and encouraged them to take urgent actions on the following recommendations:

- The strategic role of governments
- The challenges of revitalizing nuclear education by universities
- Vigorous research and maintaining high-quality training
- The benefits of collaboration and sharing best practices.

6.5.1 Benefits of international collaboration

In the above-mentioned report, the NEA suggested that the industry, research institutes and universities need to work together to coordinate efforts better to encourage the younger generation, as well as to develop and promote a programme of collaboration in nuclear education and training, and to provide a mechanism for sharing best practices in promoting nuclear courses between member countries.

International collaboration would bring benefits, such as:

- Sharing costs among different countries, since development of training systems may be too expensive for one nation
- To counter a withering pool of training resources and knowledge
- To harmonize training standards at an international level
- To push initiatives for international skills retention, as well as to attract the next generation of scientists and engineers
- To demonstrate a united global position on future nuclear technology.

Taking into consideration the recommendations from the NEA and being aware of the benefits, some common education and training efforts at international level have already started.

Before discussing those international initiatives, it is necessary to introduce one of the most important references in training: the Institute of Nuclear Power Operation (INPO).

6.5.2 Institute of Nuclear Power Operation (INPO)

Established by the nuclear power industry in December 1979, following the recommendations made by the Kemeny Commission (set up by President Jimmy Carter to investigate the accident at the Three Mile Island nuclear power plant), the Institute of Nuclear Power Operation is a not-for-profit organization headquartered in Atlanta, Georgia, USA. The mission at the INPO is to promote the highest levels of safety and reliability – to promote excellence – in the operation of nuclear electric generating plants. This mission is accomplished through the programmes of plant evaluations, training and accreditation, events analysis and information exchange and missions of technical assistance.

In 1985 was founded the National Academy for Nuclear Training (NANT), which provides training and support for nuclear power professionals. NANT evaluates individual plant and utility training programmes to identify strengths and weaknesses and recommend improvements. Selected operator and technical training programmes are accredited through the independent National Nuclear Accrediting Board.

INPO offers, through its International Program, technical assistance to its members and access to its website including important references and very detailed descriptions of specific training programmes for different job positions.

6.5.3 Common education and training efforts at international level

It is strongly recommended for those countries starting nuclear programmes to join some of the following international associations and initiatives.

International atomic energy agency (IAEA)

The International Atomic Energy Agency (IAEA) is one of the international organizations that can support training for capacity building through the following strategies:

- Support for in-house training and sustainability through the 'Train the Trainer' approach

- Promotion of networking based on Asian Nuclear Safety Network experience
- Promotion of workshops and conferences.

IAEA general training tools for capacity building are multimedia training courses, web-based knowledge sharing, tailored training sessions and workshops, the Centre for Advanced Safety Assessment Training and the promotion of bilateral and multilateral exchanges of trainees.

In 1994 was constituted the Technical Working Group on Training and Qualification (now known as 'Managing Human Resources in the Field of Nuclear Energy', TWG-MHR). This international group, with the participation of all the countries with nuclear interests, meets every two years at the IAEA's Vienna headquarters. This biennial meeting is a valuable opportunity, as it is the only such worldwide gathering on this topic. Its objectives are:

1. To exchange information on status and trends concerning NPP personnel training and qualification in Member States.
2. To recommend future IAEA activities related to NPP personnel training and qualification.
3. To review the Agency's activities in the subject areas performed in the past two years and provide recommendations to implement the IAEA programme in the next two years.

EURATOM FP-7 research and training: the need to maintain nuclear competence, EURATOM (2007)

FP7 is the short name for the Seventh Framework Programme for Research and Technological Development. This is the EU's main instrument for funding research in Europe and it will run from 2007 to 2013. FP7 is also designed to respond to Europe's employment needs, competitiveness and quality of life.

The framework programme for nuclear research and training activities will comprise Community research, technological development, international cooperation, dissemination of technical information and exploitation activities as well as training.

Sustainable Nuclear Energy Technological Platform, SNETP (2007)

The SNETP was officially launched in 2007. Today, SNETP gathers about 70 European stakeholders from industry, research and academia, technical safety organizations, non-governmental organizations and national representatives. SNETP aims to support fully through R&D programmes the role of nuclear energy in Europe's energy mix, and its contributions to the security and competitiveness of energy supply, as well as to the reduction of greenhouse gas emissions. To achieve this objective, SNETP has elaborated a Strategic Research Agenda (SRA) that identifies and prioritizes the research topics.

SNETP has set up a specific Working Group dedicated to Education, Training and Knowledge Management (ETKM) issues, with the support provided by the European Nuclear Education Network (ENEN) Association. This workforce will in part also provide qualified staff to Europe's nuclear industrial sector to accompany the development of the sector in the next decades.

World Nuclear University, WNU (2003)

To help point the way towards a globalizing nuclear profession, the World Nuclear Association has worked with the IAEA, WANO and the NEA to create the new World Nuclear University. The WNU is a partnership in which these four global organizations cooperate together, and with leading institutions of nuclear learning, in activities to enhance nuclear education and leadership for the twenty-first century. The WNU partnership is supported by a small multinational secretariat in London composed of nuclear professionals seconded by key governments and nuclear enterprises.

The flagship of the partnership is the WNU Summer Institute, an annual six-week event designed to educate and inspire an international group of young nuclear professionals who show promise as future leaders in the world of nuclear science and technology.

European Nuclear Energy Forum, ENEF (2007)

The ENEF is a unique platform for a broad discussion on transparency issues as well as the opportunities and risks of nuclear energy. Founded in 2007, ENEF gathers all relevant stakeholders in the nuclear field: governments of the 27 EU Member States, European institutions including the European Parliament and the European Economic and Social Committee, nuclear industry, electricity consumers and the civil society.

Within ENEF there exists a Working Group concerning education and training in the Permanent European Human Resources Observatory.

6.5.4 Networks of Excellence in Education and Training, NEE&T (2010)

Other initiatives to promote the renewal of competencies are on-going in different fields: nuclear safety courses organized by the Network of Excellence for Severe Accident Research (SARNET), winter and summer schools in the field of actinide science organized by the ACTINET Network of Excellence, the Frédéric Joliot and Otto Hahn Summer School on Nuclear Reactors, and the Latin American Network for Education and Training in Nuclear Technology (LANENT), currently in the process of creation, are examples of such initiatives.

Especially active are the European Nuclear Engineering Network (ENEN) and the Asian Network for Education in Nuclear Technology (ANENT).

European Nuclear Engineering Network, ENEN (NEE&T, 2010)

The ENEN association, currently comprising 41 members, plays a major role in shaping Europe's education system. ENEN facilitates exchanges and cooperation within academic institutions and strengthens their interactions with research centres. It delivers the certificate of European Master of Science in Nuclear Engineering (EMSNE). It further develops, promotes and supports ENEN exchange courses in nuclear disciplines including reactor safety, waste management and radioprotection. It facilitates and coordinates the participation of universities in European research projects.

To the benefit of the end users, ENEN preserves nuclear knowledge and improves access to expertise by developing and establishing databases, websites and distance learning tools. It has a role as an interface between academia and industry, to define, disseminate and support interesting projects and research topics for internships, masters' theses and PhDs. by developing a framework for mutual recognition of professional training, licensing and professional recruitment throughout the European Union, ENEN is creating a nuclear 'European Education and Training Area'.

Asian Network for Education in Nuclear Technology, ANENT (NEE&T, 2010)

ANENT is set up to promote, manage and preserve nuclear knowledge and to ensure the continued availability of talented and qualified human resources in the nuclear field in the Asian region and to enhance the quality of the resources for the sustainability of nuclear technology.

The objective of ANENT is to facilitate cooperation in education, related research and training in nuclear technology in the Asian region through:

- Sharing of information and materials on nuclear education and training
- Exchange of students, teachers and researchers
- Establishment of reference curricula and facilitating mutual recognition of degrees

- Serving as a facilitator for communication between ANENT member organizations and other regional and global networks.

The essential functions of ANENT are to integrate available resources for education and training in synergy with existing IAEA and other mechanisms, to create public awareness about the benefits of nuclear technology and its applications, to attract talented youth in view of alternative competing career options, to encourage senior nuclear professionals to share their experience and knowledge with the young generation, and to use information technology, in particular web-based education and training, to the maximum possible extent.

6.6 Initial and sustained training programmes

The strategy to develop and implement a comprehensive training system should be established at an early stage (around eight years before the first fuel loading) of a new nuclear programme. This strategy will take as an input the necessity of human resources according to the staff planning (number of professionals and technicians needed to incorporate each year).

The first step in this strategy will be to analyse and identify the training needs; it will be very useful to organize different groups within the organization which will need a common initial training programme, for instance managers, engineers and instructors. These strategic groups will play an important role in the future nuclear project. These initial training programmes are usually organized from the most generic (nuclear 'indoctrination' courses) to the most specific according to the different training needs. The strategy to implement the training system will constitute an important section of the documentation usually submitted during the licensing process.

The training system, including the training organization and infrastructure, must be fully completed and ready for implementation at least three years before the fuel loading in order to train the operation and maintenance plant staff. Meanwhile some initial training can be delivered using external expertise support if there is no national training organization available. External support can be provided by the vendor (within the supply contract), international training organizations or training centres.

6.6.1 Systematic Approach to Training (SAT)

According to international standards, the initial and on-going training programmes for the personnel involved in the operation of the nuclear facility must be designed and implemented following a Systematic Approach to Training (SAT) methodology. Establishing SAT at an early stage in the project will help to ensure that an effective training system is set up within the project and that those areas where training services and support can be appropriately outsourced to vendors and/or national education and training organizations are correctly specified.

A training programme must have been developed according to the following phases, if it is to fit within the SAT concept:

- Analysis
- Design
- Development
- Implementation
- Evaluation.

For all jobs that have a potential impact on the safe and reliable operation of nuclear facilities, the training needs associated with both technical competence and soft skills should be considered and analysed as part of the SAT process.

An important activity of the SAT analysis phase is job analysis. Job analysis is a method used to obtain a detailed listing of the duties and tasks of a specific job. The results of job analysis are an important input to the SAT design phase. Job analysis results are also important for other HR-related purposes, such as recruitment and selection, HR planning, training, qualification and authorization, employee development, succession planning and career development.

Once a training programme is running, new information or events can trigger training needs analysis, for example changes in regulatory requirements, plant modifications, new procedures, feedback from job incumbents, supervisors, trainees or instructors, operating experience, and weaknesses in training processes or performance deficiencies, amongst others.

The most important outcome of the design phase is the training objectives. Clear training objectives which are measurable and based on job requirements constitute the basis for designing training programmes, developing training materials and performing post-training assessments of competencies.

During this phase the different training settings (classroom, simulator, workshop, laboratory, plant for on-the-job training) and training tools, suitable for achieving the training objectives, should be identified. Training tools that are particularly important in the nuclear industry include simulators, equipment for workshops and laboratories, mock-ups, computer-based and web-based training systems, e-learning platforms, and video and audio training aids.

Finally, the standards and associated assessment methods are determined during the design phase.

The outcomes from the development phase are the suitable training materials which support the training tools, such as lesson plans, student handouts, simulator scenarios, workshop and laboratory practices and on-the-job training guides. Particularly important activities within the development phase are the training of instructors and validating training materials during a pilot course to ensure the required quality of training delivery.

It is during the implementation phase when training is conducted in the different training settings. If the analysis phase has been well done, only relevant training will be delivered. The implementation phase also includes an assessment of whether students have achieved the standards identified in the training objectives. The assessment of competencies should lead to a formal process of qualification and authorization of personnel to work in an efficient and safe manner without direct supervision.

Training evaluation is one of the most important phases to guarantee the effectiveness of training programmes and improve performance. According to Kirkpatrick (2011) four levels of evaluation can be used to determine the impact of training:

- Level 1: Participants' reactions to the training
- Level 2: Participants' achievement of training objectives
- Level 3: Transfer of competencies acquired through training to job performance or behaviour
- Level 4: Impact of training on organizational performance.

The conclusions from the evaluation are used as feedback for the rest of the SAT phases for training improvement.

6.6.2 Training system elements

A comprehensive training system should include full and accurate descriptions of the following elements.

Training regulatory requirements

It is necessary to identify the national and international standards for training that are applicable and the training 'certification' model according to the training regulatory requirements. It is essential to decide the roles played by the plant management, the training organization, the regulator or, if appropriate, an external independent organization in the 'certification', 'accreditation' or any other concept that will assure the quality of the training programmes.

A good example of an external certification model is the training accreditation process conducted by the National Academy for Nuclear Training of INPO.

Training organization, management and staffing

An important consideration when organizing a training system is to clearly define the role of the line managers, as they are the owners of the training programme of their people, the role of the training manager, as the training consultant and administrator, and the responsibilities for training and qualification of the line managers, training managers and plant personnel.

The training organization should be designed so as to include the functions and responsibilities of instructors and other training staff and the description of the training committees (expected attendees, meeting frequency and proposed agenda). The training organization manual has to include a detailed description of the administration of training and qualification activities. Finally, it is recommended to identify some indicators of training effectiveness.

Training quality plan and procedures

The training process needs to be well documented following international training quality standards. It will be necessary, therefore, to develop a minimum set of training procedures relating to:

- Training needs analysis
- Training programme design
- Training material development
- Exam development
- Delivering training sessions
- Trainee performance evaluation
- Instructor training and qualification
- Training system effectiveness evaluation.

Training programmes and materials

The development of the training materials (instructor guidelines, student handouts or training aids) is one of the activities that require most time and resources.

Training programmes include proposed long-range training schedules for each programme and a description of how these training schedules will be updated and maintained.

The documentation needed for plant operation and maintenance and design updating needs to be included in the vendor's scope of supply. This documentation should be structured in such a way as to facilitate effective knowledge transfer to be included in training materials.

Instructors

It is very important to define the recruitment sources, selection criteria and training programmes for instructors and subject matter experts. Usually instructors need to demonstrate proficiency and experience in the field they are going to teach and to be trained on a specific training programme for trainers, in order to acquire the pedagogical skills needed to be able to put into practice the training quality assurance programme.

At this stage it is necessary to have:

- The instructor training programme description for initial and continuing training
- The initial and continuing training requirements for on-the-job trainer and task performance evaluator qualification (or the references to the procedure requirements)
- The process for maintaining instructors' technical knowledge and proficiency
- Guidelines for the observations of instructor performance.

Training facilities, training tools and simulators

Finally, it is necessary to design the facilities and equipment at the training centre. The description of the training resources will include buildings, classrooms, laboratories, simulators, mock-ups and other training delivery settings and equipment.

Among the training tools the simulator deserves special attention. According to IAEA (2009b) a key lesson learned regarding commissioning of a nuclear facility is the importance of having a plant-referenced, full-scope control room simulator available well in advance of nuclear facility operation. This simulator not only provides a unique tool for training nuclear facility control room personnel, but also is important for tasks such as normal, abnormal and emergency operating procedure development and validation, development and validation of commissioning tests, validation of digital control systems, and training of other plant personnel.

For many new nuclear facility projects, a full-scope simulator is provided as part of supplying the nuclear facility package. Integrating the simulator development and training schedule with the overall commissioning schedule is very important. According to the US Nuclear Regulatory Commission, the simulator should be ready for training three years before the fuel load.

Important considerations regarding simulator information are a simulator configuration control process description; a plan for acquiring, validating and using a plant reference simulator (or if a plant reference simulator is not yet available, a description of how and when a part-scope or non-plant-referenced simulator will be used during the training and how and when that simulator will become plant-reference); a list of unresolved simulator deficiencies and recent simulator fidelity data; and simulator performance indicators or process description.

6.6.3 Training for human performance improvement

Appropriate attitudes on the part of nuclear facility personnel have to be ensured. Due attention should be paid to the fact that the required attitudes cannot be achieved only through education and training. Attitudes also depend on individual characteristics and organizational culture. The behaviour of nuclear facility managers and their ability to be everyday role models for their personnel are crucial factors.

Managers of nuclear facilities should embrace their roles in evaluating training to improve its effectiveness and to improve performance, in the same way as they embrace performance improvement. They are responsible for the behaviour of their employees and for the consequences of that behaviour. Following training, managers should observe the performance of their recently trained employees, provide timely, behaviour-specific feedback to those employees, evaluate the impact on organizational performance, and provide feedback to the trainers so that they can improve the quality of the training.

As indicated by the National Academy for Nuclear Training (2002) in their document ACAD 02-004, the training organization can support line managers and encourage professionalism through activities such as the following:

- Provide training that improves station and personnel performance.
- Ensure that the training staff serve as a role model for other personnel by exhibiting a high level of professionalism while conducting training in every setting.
- Ensure that training personnel model the standards and expectations held by the line managers; conduct training according to clear standards of performance and behaviour; let personnel know what is expected of them and hold them accountable; emphasize pride of ownership and accountability; and clarify the method for performing tasks correctly.
- Provide input to managers to encourage them to recognize exceptional personnel performance during training. Likewise, recognize superior instructor performance. Course completion certificates and awards can foster a sense of training and qualification accomplishment in personnel and their instructors.

Training should be considered a strategic tool to foster human performance excellence and therefore the improvement of safety and reliable and efficient plant operation.

6.7 Sources of further information and advice

This section introduces some relevant training issues which enlarge the information supplied in the chapter.

ENTRAC: Electronic Nuclear Training Catalogue

The IAEA has developed the Electronic Nuclear Training Catalogue (ENTRAC) which provides a method for gathering, sharing and maintaining training information and materials. The Internet site http://entrac.iaea.org (accessed 22 May 2010) provides access, after registration, to information and documents on personnel training, to the related IAEA technical documents, and also to the documents and data the Member States provide to IAEA for information exchange and sharing.

The IAEA has also established a programme to support the development of standardized training packages and distance learning tools in the field of nuclear safety. Information on this programme and the training materials available can be found at http://www-ns.iaea.org/training/ (accessed 22 May 2010).

SAT: Systematic Approach to Training

Further detailed information regarding SAT methodology can be found in the Institute of Nuclear Power Operations (1993), *Principles of Training System Development Manual*, ACAD 85-006, as well as several publications from the International Atomic Energy Agency: IAEA (2000), *Training solutions / Analysis phase of systematic approach to training (SAT) for nuclear plant personnel*, IAEA-TECDOC-1170, and IAEA (2001), *A systematic approach to human performance improvement in nuclear power plants,* IAEA-TECDOC-1204.

ACAD: National Academy for Nuclear Training

Information on the US accreditation process is detailed in National Academy for Nuclear Training (2003), *The objectives and criteria for accreditation of training in the nuclear power industry,* ACAD 02-001; National Academy for Nuclear Training (2003), *The process for accreditation of training in the nuclear power industry*, ACAD 02-002; and for new plants, National Academy for Nuclear Training (2008), *The process for initial accreditation of training in the nuclear power industry*, ACAD 08-001.

The National Academy for Nuclear Training (2002), *Guidelines for the conduct of training and qualification activities*, ACAD 02-004, contains exhaustive information about the organization, responsibilities and operation of training centres.

6.8 References

CFEN (2008), French Council for Education and Training in Nuclear energy. Available from http://www.cfenf.fr (accessed 13 August 2010).

Clean and Safe Energy Casenergy Coalition (2009), *Job Creation in the Nuclear Renaissance,* Casenergy Coalition, updated April 2009.

ENEF (2007), European Nuclear Energy Forum. Available from http://ec.europa.eu/energy/nuclear/forum/forum_en.htm (accessed 22 May 2010).

EURATOM (2007), FP7: Framework 2007–11. Available from http://ec.europa.eu/research/fp7/index_en.cfm?pg=euratom (accessed 22 May 2010).

GoNERI (2007), University of Tokyo Global COE Program: Nuclear Education and Research Initiative. Available from http://www.n.t.u-tokyo.ac.jp/gcoe/index.html (accessed 13 August 2010).

International Atomic Energy Agency (1980), *Manpower development for nuclear power, a guidebook*, IAEA Technical Reports Series no. 200, Vienna.

International Atomic Energy Agency (2004), *Nuclear power industry ageing workforce: Transfer of knowledge to the next generation*, IAEA-TECDOC-1399, Vienna.

International Atomic Energy Agency (2007a), *Managing the first nuclear power plant project*, IAEA-TECDOC-1555, Vienna.

International Atomic Energy Agency (2007b), *Milestones in the development of a national infrastructure for nuclear power*, IAEA-NG-G-3.1, Vienna.

International Atomic Energy Agency (2007c), *Considerations to launch a nuclear power programme*, Vienna.

International Atomic Energy Agency (2008a), *Human resource issues related to an expanding nuclear power programme*, IAEA-TECDOC-1501, Vienna.

International Atomic Energy Agency (2008b), *Decommissioning of nuclear facilities: Training and human resource considerations*, IAEA Nuclear Energy Series no. NG-T-2.3, Vienna.

International Atomic Energy Agency (2009a), *Workforce planning for new nuclear power programmes*, IAEA- NG-T-3.3 Draft, Vienna.

International Atomic Energy Agency (2009b), *Managing human resources in the field of nuclear energy*, IAEA Nuclear Energy Series no. NG-G-2.1, Vienna.

Kirkpatrick, D. (2011) *The Kirkpatrick Model*. Available from http://www.kirkpatrickpartners.com (accessed 12 September 2011).

NANT (2002), National Academy for Nuclear Training, *Guidelines for the conduct of training and qualification activities*, ACAD 02-004.

NEE&T (2010), Networks of Excellence in Education and Training:

- Network of Excellence on Severe Accident Research (SARNET). Available from http://www.sar-net.org (accessed 22 May 2010).
- Network of Excellence on Fuel Cycle Research (ACTINET). Available from http://www.actinet-network.org/ (accessed 22 May 2010).
- Frédéric Joliot / Otto Hahn Summer School on Nuclear Reactors. Available from http://hikwww4.fzk.de/fjohss/ (accessed 22 May 2010).

- European Nuclear Education Network (ENEN). Available from http://www.enen-assoc.org/ (accessed 22 May 2010).
- Asian Network for Education in Nuclear Technology (ANENT). Available from http://www.anent-iaea.org/ (accessed 22 May 2010).

NEUP (2009), Nuclear Energy University Programs. Available from https://inlportal.inl.gov/portal/server.pt/community/neup_home/600/home (accessed 22 May 2010).

NTEC (2008), Nuclear Technology Education Consortium. Available from http://www.ntec.ac.uk/ (accessed 22 May 2010).

Nuclear Energy Agency (2000), *Nuclear education and training. Cause for concern?*, AEN/NEA, Paris.

SNETP (2007), The Sustainable Nuclear Energy Technology Platform. Available from http://www.snetp.eu/ (accessed 22 May 2010).

WNU (2003), World Nuclear University. Available from http://www.world-nuclear-university.org/ (accessed 22 May 2010).

7 National technical capability development in nuclear power programmes

S. K. SHARMA, formerly of Atomic Energy Regulatory Board of India, India

Abstract: This chapter briefly describes the need and the means for developing national technical capabilities for setting up the first nuclear power plant (NPP) in a country and its operation and management in the long term. Orientation training of staff in nuclear science and technology, including training in a research reactor, development of a technical core group and national participation in siting and construction of the NPP, is recommended for the initial phase. For the longer term, enhancement of expertise in areas such as reactor core management, in-service inspection and management of radioactive waste and spent fuel have been suggested. The importance of developing technical support organizations, developing national safety standards and participation in international cooperative activities is also touched upon.

Key words: technical capabilities for establishing first nuclear power plant (NPP) in a country; national participation in siting and construction of NPP; preparing for start of NPP operation; long-term operation and management of NPP.

7.1 Introduction

A nuclear power plant (NPP) is a complex machine which requires personnel with expertise in a number of technical areas for its siting, construction, commissioning, operation and management in the longer term. Also, establishing a NPP entails national commitment to safety for a very long period. That would include the NPP operating lifetime, its safe keeping after cessation of operation and the time required to carry out its decommissioning. This period could extend to 100 years or even more. It would be neither possible nor practicable to depend entirely on technical support from the reactor vendor or other agencies outside the country for such a long period. For this reason, it is not only desirable but absolutely essential that requisite national technical capabilities be initially developed for establishing the first NPP in the country and these be progressively augmented towards managing the nuclear power programme and its likely future expansion.

Establishing the foundation for technical development and using a research reactor as stepping stone are discussed in Section 7.2. Section 7.3

describes the need for obtaining a good understanding of the NPP design in the operating organization, the regulatory body and the technical support organizations. Sections 7.4 and 7.5 present the advantages of national participation in siting and in design, equipment manufacture and construction of the NPP respectively. Technical capabilities required for commissioning the NPP and for its safe and efficient operation are described in Sections 7.6 and 7.7 respectively. Areas in which building of national technical competence for long-term operation of the NPP and for future expansion of nuclear power programme in the country is necessary are detailed in Section 7.8. Decommissioning aspects are covered in Section 7.9 and some sources of further information on the topics of the chapter are listed in Section 7.10.

7.2 Establishing the foundation for national technical development

For servicing a nuclear power programme over an extended period of time, it would be essential to have national capabilities in a number of technical and operational management areas beyond conventional engineering. Some of the technical areas are reactor physics, reactor chemistry, radiation protection, management of reactor core, management of spent fuel, reactor control and management of radioactive waste arising from reactor operation. Examples of operational management areas are development of technical specifications for operation and operational procedures including those for upset conditions and for accident management, configuration control of the plant and various administrative procedures for round-the-clock operation of the NPP. Technical capabilities are also necessary for carrying out thermal hydraulic analysis, ageing assessment of systems, structures and components and probabilistic safety assessment. National capabilities in these fields can be developed by getting personnel trained abroad in theory as well as in operation of a nuclear power plant of a design similar to the one envisaged to be established in the country. A good method could be to begin with setting up a research reactor and getting personnel trained in this facility first.

7.2.1 Research reactor as a stepping stone for nuclear power

A research reactor with a thermal power rating of a few megawatts has all the systems of a power reactor except those that are related to raising steam and operating the turbine generator for producing electricity. Therefore a research reactor would serve well for personnel to obtain a good understanding of the intricacies and complexities of controlling the fission chain reaction and the overall operational management of a nuclear reactor.

Training in the conventional engineering part of a NPP can be imparted in large-sized fossil-fuelled electricity generating plants in the country. A research reactor also forms a nucleus around which several scientific and engineering laboratories get established to create a multidisciplinary research centre. Such a centre would then serve as the nodal technical organization to support the nuclear power programme in the long run.

The experience gained in design, construction and operation of a research reactor is extremely helpful in developing a sound foundation for the nuclear power programme. The personnel trained in a research reactor are able to quickly assimilate the knowledge required for operating a NPP and thus a good cadre of well-trained personnel can be created in a reasonably short time for managing the nuclear power programme in the country. Operating a research reactor also gives a boost for establishing the safety culture that is so essential for the success of the future nuclear power programme. It is perhaps for this reason that all countries operating nuclear power plants today started their nuclear activities by first establishing a research reactor.

A research reactor facilitates production of radioisotopes that are extensively used in medical, industrial and other applications. This provides a good opportunity for establishing facilities for preparing targets for irradiation in the reactor, processing of the irradiated materials for producing sealed sources and radiopharmaceuticals, transportation of radioisotopes and their various medical and industrial applications for societal benefit. In a way this beneficial aspect of nuclear energy helps in conditioning the public mind towards acceptance of nuclear power subsequently.

Personnel with experience in operation and management of a research reactor can be readily inducted in the regulatory functions and this helps in early establishment of the regulatory body with competent staff.

7.2.2 Induction training of human resources

A large number of personnel trained in a variety of fields are required to support a nuclear power programme in the long term. An initial training programme is necessary to orient the newly inducted persons in nuclear science and technology where after they can receive advanced training. This initial training can be imparted through undergraduate nuclear engineering courses. However, after enrolment of students in such courses it would typically take four to five years before they become available for deployment in the nuclear power programme. An alternative could be to design capsule courses of about one year's duration where graduates or postgraduates in science and engineering could be taught nuclear subjects. Such courses are available in a few countries and arrangements may be made for getting personnel trained in these centres. However, subsequently it would be advantageous to establish such a course within the country to be able to train a larger number of personnel on a regular basis and at low cost. For personnel who are going to be engaged in the operation and maintenance (O&M) functions of the NPP, further on-the-job training should be arranged in an operating NPP or a research reactor such that they can be readily inducted into the detailed O&M training for the NPP to be constructed to become licensed operators at the earliest opportunity.

7.2.3 The technical core group

It is advisable to induct a small group of carefully selected personnel who have a few years of experience in conventional industry and put them through the nuclear orientation training as well as practical training in a research reactor or in a NPP. These personnel will form the technical core group for initiating the nuclear power programme in the country. This group can assist in drawing up the design specifications for the first NPP to be set up, interacting with international reactor vendors and finalizing the type and size of the NPP to be installed. Personnel of this core group will become the team leaders for the key activities during construction, commissioning and operation of the NPP. They will in turn train their younger colleagues and thus help in creating a large cadre of well-groomed experts to manage the first NPP and the future expansion of nuclear power in the country.

7.2.4 Work discipline and safety culture

It is of utmost importance that right from the beginning a strong emphasis is laid on formal training and having well-formulated procedures in place for conduct of all activities. Further, quality assurance and careful attention to safety including industrial safety must be made an essential part of all work. These elements will aid in developing a strong work discipline and safety culture in the operating organization, as also in the regulatory body and the technical support organizations, which is so essential for the success of the nuclear power programme in the long run.

7.3 Understanding the nuclear power plant (NPP) design

A general understanding of the various designs of NPPs available like the boiling water reactor, the pressurized water reactor and the pressurized heavy water reactor needs to be obtained by national experts initially. This would help them appreciate the characteristics of each design in respect of capital cost, construction time, operability, safety, integration of the NPP in the electricity grid, requirement of manpower for operation, fuel requirements and management of spent fuel and radioactive waste arising from operation. This understanding will enable them to have a proper interaction with prospective reactor vendors and help in the selection of the first NPP to be installed. It will also help them in explaining the justification for starting the nuclear power programme in the country to the public and the media.

After the decision on the NPP to be installed is made, a detailed study of the design must be done by the personnel in the operating organization, the regulatory body and the technical support organizations. It is extremely important to obtain a sound understanding of the design, not only for the operation of the NPP but also for the safety reviews to be conducted by the regulatory body before the various licensing stages of the NPP, viz. construction, commissioning and operation. Subsequently this understanding will be of great help in the effective and efficient regulation of the NPP during its operational lifetime. The construction group personnel should also learn the basic design of the NPP so as to be able to clearly understand and appreciate the need for maintaining high quality standards during construction.

During the review of the preliminary safety analysis report of the NPP design by the regulatory body, a number of questions will be raised and several clarifications will have to be obtained. In order to ensure that the queries are pertinent and focused, the regulatory body must have a good understanding of the design. In the absence of such understanding many trivial issues may get overemphasized that will result in loss of valuable time and the real issues getting eclipsed. It may also create a strained relationship between the regulatory body and the operating organization, leading to generation of a tendency in the operating organization to hide information or to submit only the minimum required information. Such tendencies may undermine the very purpose of conducting the safety reviews. Mutual trust and professional respect between the operating organization and the regulatory body are essential for the proper and smooth conduct of the licensing process.

On the part of the operating organization it is essential that they 'own' the design such that the need for referring questions to the reactor designer is minimized. This is possible only when the design and the design basis are well understood and well appreciated by the operating organization. Such understanding of the design is possible only through an elaborate training of the operating organization as well as the regulatory body personnel that needs to be arranged by the NPP vendor. The training should also include hands-on operation training in a NPP of a similar design and the operating experience feedback from NPPs of similar design as also the applicable experience from NPPs of other designs.

The understanding of the design should be further improved during the commissioning of the NPP as this stage provides a unique opportunity for obtaining deeper insights of the design during testing of individual components and the integrated testing of systems. A sound understanding of the NPP design so developed will not only make the safety review process more effective and efficient but also be invaluable during the longer-term operation of the NPP.

7.4 National participation in siting

After the decision on the design and capacity of the first NPP to be installed in the country is made, an appropriate site for the NPP is to be selected. For this purpose it is advisable to first identify a few candidate sites that meet the basic criteria for setting up a NPP. This can be done by a team of national experts who have the knowledge and experience in similar work performed earlier for locating thermal power plants, hydroelectric power stations and other conventional industries. Expertise in specific scientific fields related to siting is also likely to be available in various national scientific and academic institutions, and personnel from such institutions should be appropriately included in the work. It may still be necessary to include a few experts from outside and if necessary the report of the national team may be subjected to a peer review. However, it is essential that national expertise in all relevant areas for site selection be developed at the earliest. This can possibly be done during the time when the document detailing the design requirements of the proposed NPP that defines its technical parameters including its power rating is being developed.

The regulatory body should also obtain the required technical know-how for safety evaluation of the proposed site at an early date and should develop a core group for the purpose. This group will carry out the initial safety evaluation of siting proposals and provide support to the expert committee constituted by the regulatory body to perform the detailed safety evaluation for consideration of licensing of the site.

7.4.1 Siting criteria

While screening criteria available internationally can be made use of for deciding on the NPP site from amongst the candidate locations, there will be several local considerations such as land use and water use around the site, the proximity of the site to heritage buildings or archeological monuments, and the likely extent of displacement of local population and its social consequences that need to be taken into account. Apart from screening criteria there are several desirable criteria such as ready availability of access roads to the site, infrastructure available nearby to facilitate construction and the existence of sea port and railhead nearby for transportation of heavy and large components to the site. For these reasons it is essential that strong national participation be ensured in the selection of the site for the NPP.

7.4.2 National activities

After consideration of the above criteria, the selected site has to be checked for its engineerability to meet the safety requirements, given its characteristics such as its seismicity, geology, hydrology, soil characteristics and vulnerability to flooding. These assessments can be made by local experts with appropriate outside support where required. Another consideration in site selection should address the capability of the site to host future NPPs. The reason is that worldwide it is now recognized that it is advantageous to install several NPP units at one site, the 'cluster concept' as it is called. This concept facilitates better utilization of the infrastructure developed at the site including the trained manpower available readily. While doing this, due consideration has to be given to factors like the adequacy of the ultimate heat sink, sharing of systems between the units, and feasibility of construction of new units with one or more units in operation at the site. Security implications of the presence of a large construction force including contractor personnel at the site with operating units in existence also need to be addressed. One other benefit of the cluster concept is that a nuclear training centre including a training simulator for NPP units of the same design can be established at the site to cater to the manpower training requirements. Experienced personnel from the operating units who will be readily available to impart training to newcomers will be another advantage for the functioning of the training centre at such a site.

The site should also be checked from the consideration of storage and disposal of radioactive waste that will be generated from the operation of the NPP. In case it is planned to have the waste repository at a different location, it should be ensured that temporary storage of the waste at the site is feasible before it is shipped out.

The radiation dose to the public, by both direct as well as indirect exposure pathways, should be ensured to be well within the prescribed limits. Appropriate apportionment of the committed radiation dose to the public for the first NPP unit should be done, keeping sufficient reserve for future units that are planned to be installed at the site. The site should also be amenable for implementation of countermeasures that may be required in the unlikely event of an accident with significant impact in the public domain.

A detailed radiological survey of the environment around the site should be carried out well before the start of the NPP operation towards establishing the background radiation levels. These surveys should then be carried out periodically after the NPP goes into operation to assess the radiological impact of plant operation on the site. It is useful to establish an environmental survey laboratory for this purpose. Such surveys involve measuring very low background radiation levels and extremely low levels of radioactivity in samples of soil, air, water, vegetation and food items. To carry out such measurements a good deal of expertise using sophisticated instruments is required and the instruments have also to be calibrated periodically using standards. Towards ensuring correctness of measurements a good practice is to engage in intercomparison exercises with other laboratories carrying out similar work. As the environmental survey work starts before the NPP is established and continues throughout the operating life of the NPP and beyond, it is important that national expertise in this field is developed early and maintained at the state-of-the-art level.

The work done for the siting of the first NPP should be utilized to further augment the expertise in this field in the operating organization and the regulatory body as well as the technical support organizations, taking into account new technological developments and worldwide experience in siting. This will be of immense use in siting future NPPs as also during periodic safety review of operating units towards ensuring that the site continues to meet the current siting criteria.

7.5 National participation in design, equipment manufacture and construction

7.5.1 Plant design and equipment manufacture

A NPP consists of the nuclear steam supply system (NSSS) and the balance of plant (BoP). The NSSS comprises the reactor core and all structures, systems and components (SSCs) required for controlling the reactor power, shutting down the reactor when required and maintaining it in a safe shutdown state. The other SSCs of the NSSS are those that are required for cooling the reactor core during the operating as well as shutdown state and for containment of radioactivity during normal and off-normal operating conditions, including accident conditions. Design, manufacture and construction of the SSCs of the NSSS have to meet stringent nuclear standards that require a great deal of specialized expertise and experience. For this reason it is unlikely that the industry in an emerging nuclear power country can undertake this work.

The BoP comprises SSCs that are also found in conventional industry. It may therefore be possible for local design and manufacturing organizations to undertake some of this work. It is, however, to be borne in mind that many of the SSCs of the BoP are also directly or indirectly related to

NPP safety and hence have to be designed, manufactured and tested to high standards. It would be advisable that a careful survey is done in consultation with the reactor vendor to establish the feasibility of entrusting specific tasks to the local industry. For some of the tasks the capability of the local industry may have to be suitably augmented. All efforts should, however, be made to maximize the participation of national experts and manufacturing industries with a clear agreement with the reactor vendor. Such participation is very useful for enhancing national capabilities for supporting the future expansion of a nuclear power programme and for undertaking more complex tasks in future, including those related possibly to the NSSS also.

As stated earlier, all NPP equipment and components are to be manufactured to meet stringent standards. This makes quality assurance (QA) an important part of their manufacture irrespective of the manufacture being done by local or any foreign industry. The NPP owner should therefore establish early the capabilities and means for ensuring QA during various identified stages of manufacture of all components. The regulatory body should have its own independent capabilities and system in place for carrying out inspection during manufacture for QA. This can be achieved only if a sound infrastructure for QA is available in the country well before any manufacturing activity starts. While some of the QA-related tasks can be outsourced, it is essential that the plant owner as well as the regulatory body have their own technical core groups for reviewing such work.

7.5.2 Plant construction

Most of the construction activities for a NPP such as excavation, civil construction, laying of piping, cables and instrument tubing, installation of electrical, air conditioning and ventilation equipment, erection of equipment like pumps, compressors, valves, diesel generators, transformers, switchgear and the turbine generator and its associated equipment are similar to those performed in conventional industries. It should therefore be possible to identify local agencies to carry out these jobs. However, it needs to be noted that the nuclear industry is characterized by stringent quality standards and hence the contracting agencies selected should be capable of performing construction work that meets these standards. The bidding companies should be prequalified and shortlisted based on their work experience, quality of work performed earlier, availability of qualified staff in requisite numbers, and capability to mobilize the required construction machinery and manpower to complete the work according to the schedule. The successful bidders may then be selected from the organizations so shortlisted. There is the modern practice of awarding mega-contracts comprising several packages to a single construction contractor.

This is towards completing the construction work in the minimum possible time and to minimize paperwork. It would be ideal if agencies for awarding mega-contracts can be identified in the local market. If this is not possible, participation of local sub-contractors under the mega-contract should be ensured to the maximum extent possible. This will not only reduce the cost of construction but also groom the local contractors to take up future NPP construction work in the country. While national participation in construction to the maximum extent possible is highly desirable, there are certain specialized jobs such as the erection of the reactor vessel, primary coolant system piping and reactor control and protection system components that may have to be necessarily performed by experienced vendor personnel. Participation of utility personnel and local contractors in such jobs should be encouraged to the extent possible such that they can utilize this experience subsequently in commissioning and O&M of the NPP and in similar construction activities of future NPPs.

7.5.3 Construction quality

Towards ensuring high quality in construction, each piece of work must be carried out according to detailed procedures that are made available in advance. There should be an independent quality assurance agency with good participation of utility personnel to carry out quality checks at pre-identified stages or hold points. There should be formal procedures in place to deal with non-conformances from approved construction specifications, drawings and procedures, and the hierarchical levels at which their disposition is to be decided shall be identified in advance. It is most appropriate to establish a formal mechanism for communicating such changes to the commissioning group and the O&M group and to associate them in the NPP construction activities to the extent feasible. This helps in making them thoroughly familiar with the as-constructed plant. Also it is frequently necessary to make mid-course changes in construction on account of factors such as unexpected interferences encountered while laying piping and cables or non-availability of specified materials. The O&M group must properly assess the impact of such modifications and the need for modifying operating and maintenance procedures or carrying out additional checks during commissioning.

7.5.4 Regulatory inspections

The regulatory body should carry out its own independent checks during the construction phase through regulatory inspections using formal procedures. The interaction of the regulatory body during such inspections should be with the responsible utility personnel who can be assisted by the reactor vendor staff or contractor personnel. Shortcomings identified during regulatory inspections should be categorized according to their safety significance and corrective actions taken in the timeframe agreed upon between utility and regulatory body.

All deviations from design in construction are to be documented and drawings to be modified to reflect the as-constructed plant correctly. The reasons for modifications together with the justification for their acceptance and the regulatory body's consent for them should be properly documented for future reference. The regulatory inspections must also confirm that for individual systems and major equipment the construction group provides a construction completion certificate before these are taken up for commissioning.

7.5.5 Industrial safety

Adequate attention to industrial safety, including fire safety and housekeeping, is a must during the construction of an NPP. Any deficiency in these areas will not only be detrimental to the health and safety of the construction workers but will also dilute the safety culture at the site, which will have an adverse impact on the commissioning and O&M activities subsequently. Responsibility for industrial safety should be with utility personnel even though actual construction work is done by contractors. All jobs must be subjected to hazard analysis and appropriate procedures and personnel protective equipment requirements laid down for their execution. As the status of work keeps changing rapidly at a construction site, the supervisors responsible for industrial safety must make frequent visits to the site for an on-the-spot assessment and enforcement of safety requirements, including their augmentation where necessary.

7.6 Plant commissioning

7.6.1 Preparing for commissioning and start of operation

After the staff have undergone the initial training they should be associated with the experts of the reactor vendor in preparation of commissioning, operating and maintenance procedures and the technical specifications for operation that will include surveillance and in-service inspection schedules and administrative requirements. The O&M staff should also be involved in the process of review of such documents by the regulatory body.

A preliminary safety analysis report (PSAR) of the reactor and supporting technical documents are provided by the reactor vendor. These documents describe in detail the safety requirements laid down for the design and how the plant is able to meet these requirements under normal operating conditions, upset conditions and design-basis accident conditions. The PSAR also describes the engineered safety features and procedures for operator intervention to control the progression of beyond design-basis accidents and for mitigation of their consequences. The PSAR is a very important document not only for understanding the safety design of the plant but also for obtaining good familiarity with the behaviour of the plant under normal as well as abnormal conditions. The PSAR review together with the progressive review of the results of commissioning will be the basis for the regulatory body to issue a licence for initial fuel loading in the reactor core, first criticality of the reactor, ascension of power in stages and operation at rated power. A thorough study of the PSAR and the supporting documents by the operating staff and their participation during the review of the PSAR by the regulatory body helps in acquiring good familiarity with the design and operational safety aspects of the plant. Various modifications implemented during construction and those based on review of the commissioning results are suitably incorporated in the PSAR to produce the final SAR that correctly reflects the as-built plant.

Study of the SAR, the design and operating manuals of reactor systems and the equipment manuals and training on the reactor simulator will form the major component of the training of operating staff. The proficiency of the operating staff should then be checked through a system of getting checklists for individual systems signed by senior engineers, a plant walk-through, a written examination and an oral interview by a licensing board for their formal licensing for NPP operation.

7.6.2 Commissioning

Commissioning of the individual components followed by integrated commissioning of the reactor systems is done to confirm that they are able to perform their design-intended functions. While the main reactor systems are taken up for commissioning on completion of construction, service systems such as the compressed air system, electrical power supply system and water demineralization plant are commissioned in parallel with construction. Commissioning provides a unique opportunity to obtain deeper insights into the working of the reactor systems that is so essential to augment the knowledge acquired from study of documents such as design and operating manuals and PSAR. For this reason the operating staff should be intensely involved in the commissioning work. Results of commissioning should be formally reviewed in a senior-level commissioning review committee. Based on these reviews, necessary modifications in the plant and in the operating procedures should be made and additional surveillance and in-service inspection requirements should be identified. Operating staff must be involved in the commissioning review as it will help them acquire intimate understanding of the plant and the interaction of reactor systems with each other.

7.6.3 Start of operation

Initial fuel loading marks the start of operation of a NPP and therefore the complete operational discipline should be in force before the first fuel assembly is loaded in the reactor core. This would include establishment of the reactor operating island, implementation of security provisions, zoning of the operating island for prevention of spread of radioactive contamination, availability of licensed operating staff and availability of approved technical specifications for operation, operating and maintenance procedures, emergency operating procedures and emergency preparedness plans.

Achieving first criticality of the reactor is the first major step in NPP operation. For this the expected configuration of the reactor core including the anticipated position of control rods will be worked out in advance and all special instrumentation for reactor startup will have been commissioned. After satisfactory achievement of first criticality, power will be raised in steps with clearance from the regulatory body at every pre-decided stage. Some of the commissioning tests that can be carried out only with the reactor at power will now be done and their results reviewed by the commissioning review group and the regulatory body.

As mentioned earlier, the O&M staff and local technical services personnel of the reactor physics group, the fuel handling group and the radiation protection group should be fully involved at all stages from first criticality to operation at rated power, including direct participation in the commissioning tests made with the reactor at power.

7.7 Plant operation

Conduct of operation covers day-to-day operation of the NPP, execution of various operational, maintenance, in-service inspection and surveillance procedures, management of evacuation of electricity produced to the grid, and refuelling of the reactor core during refuelling outages. As brought out earlier, the NPP is to be operated by well-trained and licensed personnel within the operating envelope prescribed by the technical specifications for operation. In addition there should be formal procedures in place for shift turnaround, authorizing maintenance work, permitting installation or removal of bypasses on electrical circuits to facilitate maintenance work, altering the operating configuration of any reactor system, monitoring chemistry parameters, testing and surveillance of equipment, calibration of instruments, radiation monitoring and other health physics related checks.

During refuelling outage of the NPP a large number of maintenance, in-service inspection, surveillance and other activities of a specialized nature are undertaken in addition to the refuelling work. Sometimes fuel assemblies may require cleaning to remove crud from fuel clad surfaces that may hinder efficient heat transfer from fuel to coolant. This is done by shifting fuel from the reactor core to the spent fuel pool where the cleaning operation is carried out in equipment specially designed for this purpose. The cleaned fuel assemblies are then returned to the core. Availability of well-trained personnel and detailed advance planning is necessary such that all these activities can be performed in an organized and safe manner while keeping the refuelling outage duration to the minimum possible. Towards this end it is common to employ a number of contractor personnel during refuelling outages. It has to be ensured that these personnel have the requisite technical capabilities and adequate experience. They also need to be trained to work in a radiation environment following the prescribed work procedures. A well-established outage management system forms a very important part of the conduct of operation of the NPP. For handling fresh fuel, refuelling work, fuel cleaning operation and management of spent fuel, a separate fuel handling crew should be organized. However, the shift staff should also be trained to carry out these tasks as some of these activities may have to be performed in round-the-clock shifts.

During operation of the NPP some expected as well as unexpected operational occurrences or incidents are likely to take place. These could be due to internal causes like equipment malfunction, operator error and inadequate procedures or due to external factors like earthquakes, external flooding and disturbances in the electricity grid. Some such incidents may be safety related and some may result in a reactor trip. All such incidents should be reported in a formal manner and subjected to detailed analysis to identify the causes, including the root cause of the occurrence, and to implement the corrective actions. The exercise of reporting and analysing such incidents is by itself a good means of improving technical competence as it involves an in-depth look into the plant hardware and procedures. For the same reason personnel should also be encouraged to review the O&M experience and record this through writing reports and technical papers.

The in-house orientation training and the training provided to staff by the reactor vendor and other external agencies will suffice only for the routine operation of the reactor. Extensive support from the vendor will still be required to tackle non-routine problems as also for special jobs like working out the reactor refuelling scheme. It would therefore be necessary to carry out technical development and build national competence on a continuing basis such that dependence on the vendor is progressively reduced. Development of such competence is required for carrying out plant modifications when necessary, improving O&M procedures, in-service inspection of systems, structures and components for ageing management and for investigating abnormal occurrences and deciding on corrective actions. National technical competence is also necessary for deciding on the extension of the NPP operation beyond the initially licensed period as also on its final shutdown and eventual decommissioning. Apart from the utility, building of technical competence is necessary in the regulatory body for efficient and effective safety regulation of the NPP and in the technical support organizations to be able to provide necessary assistance by way of research, analysis and experimental support. The final aim should be to progressively attain a level of technical development so as to be able to undertake design and construction of new NPPs indigenously.

The major activities to be performed and managed by the operating organization for the safe and efficient operation of the NPP relate to management of the reactor core, maintenance, in-service inspection, reactor chemistry, radiation protection, radioactive waste management, spent fuel management, emergency preparedness and operational safety review. Important aspects of developing a sound infrastructure and a high level of national expertise necessary for the proper conduct of these activities are described in the following paragraphs.

7.7.1 Management of reactor core

Reactor core management starts with working out the scheme for initial fuel loading in the core with control rods in appropriate positions and the required concentration of neutron poison in the reactor moderator to ensure that the prescribed level of sub-criticality is maintained all the time. The next step is to compute the core configuration for its first criticality and to prepare the procedure for achieving first criticality. Special instrumentation is installed in the core for measuring the very low neutron flux in the core during the approach to first criticality. It may also be necessary to install a neutron source to ensure that the reactor startup instrumentation is comfortably on-scale before starting to reduce the boron concentration in the moderator water or withdrawing control rods for making the reactor critical.

After the reactor is made critical, the predicted and the actual core configuration for criticality are compared and any significant differences or anomalies are resolved. Thereafter various tests are conducted with the reactor at low power, such as measurement of reactivity worth of control rods and various coefficients of reactivity. Other checks such as measuring neutron flux at different axial and radial locations in the core, establishing the relation between reactor thermal power and neutron power, effectiveness of radiation shielding and response of the NPP control system during situations like partial or total load throw-off are carried out at different power levels during the reactor power ascension stages.

Thermal hydraulics computations are done to assess the power delivered from fuel to coolant and various important thermal parameters such as the fuel rod linear power, the fuel centre temperature, the fuel clad surface temperature and the temperature gradient across the fuel thickness, both for steady-state condition and under transients such as tripping of main coolant pumps and movement of control rods. Computations using coupled neutronic and thermal hydraulic codes facilitate obtaining a better and more holistic assessment of the reactor core.

After fuel burn-up proceeds to a level at which there will not be sufficient reactivity available to operate the reactor, the core will have to be refuelled. Refuelling is generally done by removing high burn-up fuel from the central zone of the core, moving the low burn-up fuel from the outer zone to the central zone and loading fresh fuel in the outer zone. As refuelling is done in the reactor shutdown state with the reactor vessel head open, care needs to be taken to prevent open vessel criticality of the core, especially when control rods are removed for maintenance work.

For performing the above activities a strong reactor physics group has to be developed at the NPP with proficiency in use of neutronic and thermal hydraulic computational codes and a good understanding of the reactor core behaviour. Based on the results of various measurements and operating experience, the computational codes will have to be fine-tuned or upgraded. This group will advise the plant management on the refuelling of the core and will work out the detailed refuelling scheme. For reactor designs with on-power refuelling, this task has to be performed on a day-to-day basis. This group will also analyse reactivity anomalies and other reactivity-related events as and when they occur and advise the plant management on the corrective actions. As the neutronic and thermal hydraulic behaviour of the core is one of the most important areas of reactor safety, the staff of the regulatory body also need to have proper understanding and appreciation of effective safety regulation. Many regulatory bodies have standing expert groups to advise them in these areas. Such advisory groups may comprise experts from within the regulatory body, the technical support organization and academic institutions and even personnel from the utility headquarters who are not directly involved in managing the reactor core. While assistance from the reactor vendor may be obtained during the initial few years for managing the core, it is absolutely essential that a high level of expertise in this field be progressively developed in the operating organization, the technical support organization and the regulatory body for managing the reactor in the long term as also for future expansion of the nuclear power programme.

7.7.2 Maintenance

A NPP comprises a large number of structures, systems and components and these need to be maintained in a good state of repair for safe and efficient operation. Maintenance can be largely divided into preventive, predictive and breakdown maintenance. All preventive maintenance activities should be well planned according to manufacturers' recommendations and executed by well-trained personnel. These schedules shall be suitably revised from time to time based on actual experience. Modern NPPs have sufficient redundancy for equipment and instrumentation items that are safety related or which are needed to be taken out of service for maintenance or calibration with the NPP in operation. Some of this equipment or components may be radioactively contaminated and hence will have to be decontaminated prior to maintenance work. Where this is not possible, maintenance may have to be done in shops that are equipped to handle contaminated parts. For predictive maintenance, the components have to be kept under surveillance to monitor any degradation such as by condition monitoring techniques or by trending their performance. Maintenance work is then done to prevent breakdown while in service. For certain redundant safety-related components the technical specifications for operation prescribe the allowed outage time. The plant maintenance groups should be well equipped to complete maintenance work on such items and return them to service within the permitted time to avoid forced shutdown of the NPP.

From the foregoing it can be seen that a high level of technical competence for all types of maintenance work must be developed in the plant staff. This can be achieved by getting some personnel trained in maintenance at other NPPs of similar design and by equipment manufacturers. These personnel in turn should train the larger number of maintenance personnel at plant site. For overhauling some of the equipment of a specialized nature such as the turbine generator, it may be necessary to engage the manufacturer's personnel during planned outages of the NPP. However, the overall responsibility for getting the work carried out and bringing the equipment back to service must rest with the plant personnel. Several maintenance activities are undertaken during planned outages such as the refuelling outage. The duration of such outages and consequently the plant load factor is heavily dependent on the capability of the maintenance personnel to complete the work in a timely manner while maintaining a high level of quality in the work performed. It must be remembered that a well-designed and well-operated NPP can give plant load factors in excess of 90% but this is possible only when it is maintained by personnel with a high level of technical skills and in the most professional manner.

7.7.3 In-service inspection

In addition to maintenance and surveillance checks on various equipment and instrumentation items, it is necessary to undertake special periodic in-service inspection (ISI) of certain structures and components that are critical for safety and continued operation of the NPP and are not easily replaceable. ISI results form an important input for the ageing assessment of systems, structures and components for long-term operation of the reactor. Examples of ISI are the assessment of the extent of radiation-induced embrittlement of the reactor pressure vessel through periodic examination of test coupons installed for the purpose, integrated leak rate testing of the reactor containment, periodic assessment of loss of prestressing in the reactor containment building, checking of steam generator tubes for thinning and existence of flaws, periodic examination of welds in the primary coolant pressure boundary, and checking of critical piping of the primary and secondary coolant system and steam system for loss of thickness or any fatigue-induced degradation. ISI is done by highly trained personnel using special techniques such as ultrasonic testing, eddy current testing and radiography. Some of the inspections, such as those of test coupons for reactor pressure vessel health assessment, may have to be carried out in hot cells of post-irradiation examination laboratories. Thus it is necessary to develop technical competence in the operating organization and in the technical support organization such that ISI can be done following specified standards and the inspection results can be properly interpreted to arrive at important decisions concerning long-term operation of the NPP

7.7.4 Reactor chemistry

Maintenance of good chemistry of reactor system fluids is essential for minimizing corrosion of reactor system components and generation of activation products that can give rise to high radiation fields on piping and equipment, resulting in increased radiation exposure of plant personnel. Reactor coolant and moderator water chemistry is generally maintained by circulating a part of the coolant flow through ion exchange resin beds. At times neutron poisons such as boron are added to the coolant in the form of boric acid to suppress excess reactivity. In this case the resins used need to be saturated with boron to prevent unwarranted boron removal that can give rise to reactivity gain. For the same reason dilution of borated water in the reactor system by inadvertent addition of unborated water must be prevented. Conversely, boron removal to gain reactivity in a controlled manner can be done by passing the coolant through ion exchange resins that have not been saturated in boron. The moderator system is normally vented to an inert cover gas such as helium in heavy water moderated reactors. Build-up of deuterium can take place in the moderator cover gas due to radiolytic decomposition of heavy water. This has to be kept within prescribed limits to prevent the concentration reaching explosive limits. For this purpose the cover gas has to be purified by passing over a catalytic recombiner. Similarly, in light water reactors hydrogen build-up in the reactor coolant is vented to catalytic recombiners. From time to time activation products that have deposited on the inner surfaces of system piping need to be removed to bring down radiation fields on piping. This is done by dilute chemical decontamination of the system, ensuring that base metal of the piping and other system components including cladding of fuel assemblies in the core are not subjected to any significant corrosion.

Chemistry of the secondary coolant of the reactor also has to be maintained within proper limits to ensure good health of the secondary system components such as the steam generators and the steam turbine. Appropriate chemicals are added to the system and the condensate is subjected to polishing by ion exchange resins before being pumped back into the feed water system. Deaeration of feed water is done to maintain dissolved oxygen content at very low values to minimize corrosion of secondary system inner surfaces.

It may be noted that chemistry control of reactor systems plays a vital role in minimizing corrosion and thereby helps in trouble-free operation of the NPP over long periods of time. It also helps in minimizing build-up of radiation fields on system piping and components thereby reducing radiation exposures of personnel. Proper maintenance of system chemistry is also necessary from a reactivity safety point of view. A well-trained and competent reactor chemistry group is therefore essential for safe and efficient long-term operation of the NPP. The technical competence of this group should be continually enhanced by in-house research as also by keeping abreast with the latest developments in this field worldwide. The chemistry group should also maintain close contact with academic and other relevant institutions in the country having expertise in specific areas such as corrosion and seek their assistance whenever necessary.

7.7.5 Radiation protection

Operation of a NPP will result in some radiation exposure to plant personnel as also to the public in the area in the vicinity of the NPP due to release of liquid and gaseous radioactive effluents from the plant. These exposures have to be maintained within the limits prescribed by the regulatory body and as low as reasonably achievable. This is done through design provisions whereby all radiation sources in the NPP are properly shielded and contained and by following appropriate procedures for carrying out O&M

activities. For limiting the radiation exposure of the members of the public, adequate checks are maintained for controlling the radioactive effluents from the NPP to the environment and developing appropriate models for assessing the exposure of the public from such effluents by way of direct exposure, as also by inhalation and ingestion through the terrestrial, aquatic and air routes. The design provisions and specified procedures shall take into account exposures during normal operation as also during off-normal situations including accident conditions. Towards this aim a robust radiation protection programme must be in place well before the start of NPP operation.

The radiation protection programme comprises monitoring the radiation exposure of all personnel inside the operating island and at other places such as in the waste disposal facility or in the away-from-reactor spent fuel storage area which have a potential for causing radiation exposure. Monitoring of external exposure is done by measuring the radiation dose received by radiation exposure monitoring devices such as thermoluminescent detectors or direct reading dosimeters that have to be worn by all radiation workers while in the plant. Internal exposures are monitored through bioassay samples and whole body counting of the workers. Radiation dose to the public is assessed by measuring radioactivity levels in air, water and soil samples around the plant and in food items including milk and milk products consumed by the public around the NPP and by estimating the dose using validated computational models.

In addition to monitoring of personnel exposures, radiation levels in various areas of the NPP and radioactivity levels in the fluids in the reactor systems and in the air in various plant areas are regularly checked. The water and air samples are also subjected to gamma spectrometry to identify the presence and concentration of various radionuclides to obtain information on the source of radioactivity in these fluids. All plant areas are regularly checked for radioactive contamination and various measures are taken, including the use of personnel protective equipment by workers and barricading of areas to prevent contamination of workers or spread of contamination. To limit the radiation exposure of workers during maintenance work or special operations like refuelling, their time of exposure is limited and temporary radiation shields are used to reduce the radiation level at the work spot. At times the work is performed using remote handling devices to bring down the exposure by increasing the distance between the workers and the radiation source. Fresh air masks are used to prevent internal exposure from intake of airborne radioactivity.

For effective implementation of the radiation protection programme a dedicated group of health physicists is required with a high level of competence in radiation monitoring, assessment of radioactivity levels in various matrices, control of radiation exposure of personnel and prevention of spread of radioactive contamination. Appropriate laboratory facilities are also required to be set up for carrying out all necessary measurements and analysis of samples. A high fraction of the total radiation exposure of plant workers takes place during refuelling, maintenance and in-service inspection work performed during planned outages of the NPP. The health physics personnel play a very important role in minimizing these exposures by advising plant personnel on the appropriate measures that need to be taken. For this reason, in many countries, the key health physics staff are formally authorized by the regulatory body after extensive training.

At times unplanned exposure of personnel may take place or personnel may get over-exposed due to loss of shielding, failure to follow prescribed procedures, inadequacy in the procedures or improper use of protective equipment. All such cases must be analysed in detail to identify the direct as well as the root causes to decide on the appropriate modifications in hardware and procedures to prevent recurrence. A high level of technical competence in the health physics group and radiation safety awareness in the workers is required for proper implementation of the radiation protection programme at the NPP with the aim of keeping all radiation exposures within the prescribed limits and also as low as practicable.

7.7.6 Radioactive waste management

Some radioactive waste will be generated from NPP operation in the form of liquid effluents, solid waste and gaseous effluents. The liquid effluents with low levels of radioactivity are treated using appropriate processes and recycled as far as possible, but some liquid effluents will have to be discharged to the environment. Such discharges are generally done using the dilute and disperse principle. The effluents are diluted, for example by the large quantities of condenser cooling water outlet from the NPP, and then dispersed in large water bodies like a lake, river or sea near the NPP site. Solid radioactive waste is generated from plant operation in the form of replaced components or their parts, piping sections, used filters, exhausted ion exchange resins, radioactively contaminated personnel protective wear like coveralls, gloves and caps and materials like mops used for decontamination of floors and other surfaces. The solid radioactive waste is stored in near-surface disposal facilities after volume reduction and packaging where feasible. Such facilities may be co-located with the NPP or they could be centralized facilities located elsewhere and may store radioactive waste from several installations. Some solid wastes such as ion exchange resins may require special treatment before disposal, such as fixation of radioactivity in the resin in cement or polymer matrix to prevent its leach-out during extended storage. Facilities for such special treatment have to be built as part of the NPP complex. Radioactive gaseous effluents are generated by neutron activation of air and suspended particulates and by pick-up of radioactivity by reactor ventilation air during its passage through radioactively contaminated areas. The ventilation exhaust air from the reactor building and other plant buildings having a potential for giving rise to airborne radioactivity is filtered through high-efficiency particulate filters for removal of particulate activity and, if necessary, through special filters like those made of activated charcoal for trapping radioactive iodine. It is then released through a tall stack into the atmosphere for dilution and dispersal.

It can be seen that radioactive waste management at NPP sites is an ongoing activity that requires special expertise. This function is important as it is to be ensured that radioactive waste disposal to the environment must be within the prescribed limits. Further, even within the specification limits, it should be kept as low as reasonably achievable to minimize adverse impact on the environment in the long term. This objective can be achieved only through having a dedicated radioactive waste management team with high technical competence. Ongoing research and development at the technical support organizations is also necessary towards developing improved processes for recycling of liquid waste and reducing waste volumes to the maximum extent possible.

7.7.7 Spent fuel management

Spent fuel removed from the reactor core has to be properly and safely stored for several years before it can be shipped out for reprocessing, final disposal or further storage at a different site. Spent fuel is stored under water in the spent fuel storage pool in fuel storage racks that have inbuilt high neutron-absorbing materials to ensure sufficient subcriticality. The pool water has to be circulated, cooled and purified to remove the decay heat transferred from the stored fuel to pool water and to maintain its chemistry parameters to minimize corrosion of the fuel cladding. For transportation of spent fuel from the NPP site, specially designed shielded casks are used and transportation is done after the decay heat in the fuel has come down to a level when natural convection cooling by surrounding air is sufficient to keep the fuel and fuel cladding temperature within specified limits.

If the storage capacity in the pool becomes insufficient due to inability to ship out the fuel for any reason, timely action is necessary for construction of away-from-reactor storage pools to augment the storage capacity. The away-from-reactor pools have to be built and operated in the same way as the storage pool at the reactor site. It is also possible to store spent fuel in dry storage casks or dry storage facilities after it has been cooled for a sufficiently long period. Such casks and facilities may have to store spent fuel for a fairly long time till the final disposition of the fuel is decided. Accordingly they have to be kept under proper surveillance by periodic checks on fuel clad integrity and structural integrity of the casks and the facilities. Expertise in spent fuel management over extended periods of time that can run into several decades has to be acquired by the operating organization. The technical support organizations and the regulatory body also need to develop adequate technical competence in this field.

7.7.8 Emergency preparedness

While NPPs are designed and operated with a very high level of safety, it is essential that an adequate level of preparedness is still maintained to deal with the highly unlikely situation of a reactor accident. Reactor accidents can be broadly categorized as design basis accidents (DBA) and beyond design basis accidents (BDBA). For a DBA the design provisions, including the engineered safety features such as the emergency core cooling system, and the reactor containment system, together with the actions taken by well-trained operators, should be able to contain or confine the radioactivity released from the reactor core such that there is no significant adverse impact beyond the NPP site. In the case of a BDBA that may be caused by unanticipated failure sequences or by multiple failures occurring simultaneously or due to a natural phenomenon of intensity greater than that considered in the design, there could be significant impact in the public domain. In modern NPP designs, due consideration is given to BDBAs also and provisions are made to enable the operator to control their progression and to minimize their adverse consequences.

The first step in emergency preparedness is to develop emergency operating procedures for all envisaged situations and to impart intensive training to operators for their execution when required. Extensive use should be made of the training simulator for this purpose. It should, however, be kept in mind that it is not possible to anticipate all possible emergency situations. At times the operators will have to use their ingenuity and take actions that might not have been included in the emergency operating procedures. This is possible only when the operators have a thorough understanding of plant behaviour and a high level of technical competence.

Emergency plans need to be in place for actions that are to be taken in case a reactor accident has a potential for or causes actual release of radioactivity outside the reactor containment. The actions could be in the form of countermeasures such as administration of prophylactics to prevent uptake of radioactive iodine by people, impounding food and milk, barricading of radioactively contaminated areas or even evacuation of affected or likely to be affected populations. For deciding on the type of emergency actions, their extent and the zone around the NPP where these need to be implemented, an assessment of the quantum of activity released has to be made. Further, the dispersal of the activity in the atmosphere and its deposition on the ground taking into account the prevailing weather conditions has to be estimated For the longer term the radiation dose to the public by the terrestrial and aquatic routes and through the food chain has to be computed. These assessments have to be made through analysis of a large number of air, water, soil and food samples for their radioactivity content and by using computational models for estimating the dose to the members of the public by direct exposure as also through the inhalation and ingestion routes.

Emergency preparedness involves developing the requisite technical competence for carrying out such assessments in quick time, deciding on the countermeasures to be implemented and finally executing the actions in an organized manner. As implementation of countermeasures will be done by the public authorities, it is essential to have a mechanism in place for proper and timely coordination between the NPP and the public authorities. Emergency exercises have to be carried out regularly according to the time schedule approved by the regulatory body to test the plans to be in a good state of preparedness to manage emergencies.

7.7.9 Operational safety reviews

The basic elements for safety in operation of a NPP are the ability to control the reactor power, ensuring adequate core cooling at all times and containment of radioactivity. Towards this aim the NPPs are designed using proven engineering practices and following the principles of defence in depth and adequate redundancy and diversity in safety-related components. However, in spite of the best design, situations can arise during NPP operation that were not envisaged in the design. Experience has shown that timely actions by competent operators may be able to ensure safety even during such unforeseen circumstances. A high level of technical competence in well-trained operators is therefore an absolute necessity. This can be achieved to a large extent by learning from operational safety reviews and operating experience feedback. It should be borne in mind that the primary responsibility for safety rests with the operator.

A formal mechanism for review of operational reports and operational incidents on a regular basis should be established by the operating organization. In addition the regulatory body should lay down criteria for reporting of safety-related operational occurrences. These reports should be reviewed to identify the causes, including the root causes, of the incidents and necessary corrective actions should then be implemented. While some of the actions can be implemented immediately, there will be other actions for which detailed analysis, experimental work or development of designs and procurement of materials or components may be required. For implementing such actions, a time schedule should be agreed upon between the operating organization and the regulatory body.

In addition to the operational safety reviews mentioned above, comprehensive and detailed periodic safety reviews should be conducted at intervals of about 10 years. For such reviews, a detailed report by the operating organization Is prepared and reviewed internally before its review by the regulatory body. The purpose of periodic safety reviews is to confirm that the NPP meets the current safety requirements and is also expected to continue to meet them till the next such review. The periodic safety reviews should also take into account the feedback from international operating experience, new knowledge available from research and the updated probabilistic safety analysis of the plant.

Another useful method for improving operational safety is through peer review by teams of international experts. Similarly the work of the regulatory body can also be subjected to international peer review. Such peer reviews bring in the benefit of experience from across the globe and the information on good practices followed by the operators and the regulatory bodies of other countries.

7.8 Longer-term operation and management

There are several areas in which national competence needs to be developed to be able to service the NPP for its proper upkeep during the longer-term operation and for future expansion of the nuclear power programme in the country. The longer-term operation would include possible extension of operation beyond the initially licensed operating period. In addition to building national technical capabilities in the major areas described in Section 7.7, other important areas for managing the nuclear power programme in the longer term are building human resources, developing technical support organizations, developing national safety standards and engaging in international cooperation.

7.8.1 Building human resources for the longer term

Towards building adequate human resources for the longer term and for the future expansion of the nuclear power programme in the country, there should be regular induction of fresh manpower. These personnel should be trained in the various aspects of nuclear science and technology as described in Section 7.2.2. Further training in specific fields should be provided while working in their respective areas in the country and by deputing them to institutions abroad engaged in advanced work.

At the NPP site a training centre should be established for training of personnel in O&M of the NPP. This centre should have facilities for classroom training, training using models of equipment and a training simulator. The training centre should also conduct refresher training and training for relicensing and upgrading the licenses of staff as necessary.

Experienced O&M personnel should be inducted in technical services functions such as refuelling outage planning, development of plant modification proposals and review of operational activities for possible improvements. Subsequently these experienced personnel can be engaged in the task of setting up new NPPs. Experienced O&M personnel can also be effectively utilized in carrying out the regulatory and technical support functions.

7.8.2 Development of technical support organizations

The operating organization as well as the regulatory body would require extensive technical support in a number of areas for efficient operation and effective safety regulation of the NPP. Such support would be needed to tackle problems that may arise during operation as also to obtain a proper understanding and assessment of the ageing-related degradation of systems, structures and components and to find appropriate solutions for their longer-term management. Also further analysis and experimental work may become necessary in the light of new information from research or operating experience. In addition, over a period of time the safety standards may get revised, leading to the need for implementation of safety upgrades that might need substantial engineering development. To cater to these needs requisite laboratories and engineering development facilities should be established and expertise generated for their effective functioning. Some examples of the facilities required are metallurgical laboratories for carrying out failure analysis of radioactive as well as non-radioactive components, assessment of the extent of irradiation-induced embrittlement in materials and post-irradiation examination of reactor fuel. Examples of facilities for engineering development are those required for testing of tools and procedures for complex repair and inspection jobs, environmental qualification and endurance testing of components and development of remotely operated tools.

Capabilities are also needed to carry out various studies and analyses such as on atmospheric dispersion of radioactivity under different weather conditions, analysis of ageing structures to check on their continued capacity to withstand design loads, and periodic updating of the probabilistic safety analysis for a quantitative assessment of the current safety status of the NPP. There are other areas also in reactor physics, reactor chemistry, control and instrumentation and computer-based systems where the technical support organizations will have to play a strong role in support of the NPP operation and regulatory effort.

Experience shows that total dependence on the reactor vendor for technical support over an extended time is neither feasible nor desirable. It is therefore necessary to establish technical support organizations in the country well before the start of NPP construction and to staff them with personnel who have received advanced training in specific fields. Facilities for analysis and engineering development should be set up and progressively augmented for effective functioning of the technical support organizations. Expertise available in the various academic and professional institutions in the country should also be utilized such as by awarding research projects to these institutions for specific development jobs and by inducting their experts in advisory committees and in development of national safety standards.

7.8.3 Development of national safety standards

Based on experience in design, operation and regulation of NPPs, several countries have developed their national safety standards for siting, design, construction, operation, decommissioning and quality assurance aspects of NPPs. Some international organizations have also developed safety standards for NPPs that codify the good practices followed globally. In the beginning, a country starting its nuclear power programme can adopt or utilize these available international safety standards as appropriate. However, after gaining some experience it is advisable that the country's regulatory body develops its own safety standards. To start with, the emphasis should be on developing those standards where the internationally available safety standards are found to be not directly applicable. This could be due to the specificities of the NPP design adopted or on account of local conditions such as climate, soil characteristics and expected frequency or magnitude of natural phenomena like precipitation, earthquakes, etc., that may be significantly different from those in other countries.

For developing safety standards the regulatory body can engage experts from its own staff and from the operating organization, the technical support organization and academic and professional institutions in the country. For ensuring good quality, formal mechanisms should be in place for thorough review of the draft documents before their publication. In addition to the safety standards that specify the safety requirements, supplementary documents like safety guides and safety manuals that provide details on the means to fulfil the safety requirements also need to be developed. The exercise of developing national safety standards is by itself a good means for enhancing the national technical competence.

7.8.4 International cooperation

International cooperative activities are an excellent means of learning the good practices in operation and regulation of NPPs followed in different countries. They can also help in enhancing national capabilities in design, analysis and research pertaining to a number of areas like seismic design of structures and components, thermal hydraulic analysis, probabilistic safety assessment, ageing management of NPPs, analysis of severe accidents and means for their management, radioactive waste management, decommissioning, safety of computer-based systems and operator response under challenging situations. There are several ways for using international cooperation for advancing the knowledge and technical competence of staff in the operating organization, regulatory body and technical support organization, such as through participation in coordinated research programmes and standard problem exercises organized by the International Atomic Energy Agency. Deputing staff to research centres abroad for advanced training in specific areas is another useful method.

It is well recognized that use of operating experience feedback not only is helpful for improving safety but also improves technical capability for analysing incidents to arrive at the root causes and lessons learned to make necessary improvements in hardware and procedures. There are several means for utilizing the international operating experience feedback such as by participation in the Incident Reporting System operated by the International Atomic Energy Agency and the Nuclear Energy Agency of OECD and a similar system operated by the World Association of Nuclear Operators. There are also the operating experience and other information reporting systems operated by vendors of NPPs of specific designs.

As already mentioned earlier, another dimension of international cooperation that is of very significant use for enhancing national technical competence is through international peer reviews. Examples of such peer reviews are the operational safety review and regulatory system review services offered by the International Atomic Energy Agency and the peer reviews organized by the World Association of Nuclear Operators. Peer reviews under the various international conventions such as the Convention on Nuclear Safety are also useful in this regard.

Cooperation with the regulatory bodies of other countries and participation in the forums of regulatory bodies of countries operating or constructing NPPs of similar designs is also very useful for improving technical capabilities of regulatory staff. Lastly, participation of staff in international conferences related to design, operation and safety of NPPs and in workshops on specific topics will also help in enhancing their technical competence by way of learning the latest developments around the globe and exchange of information with international experts.

7.9 Decommissioning

After the NPP has operated for its licensed period it will have to be finally shut down and then decommissioned. Decommissioning may also become necessary before the end of the licensed period for other reasons such as governmental decision or the operation of the plant being no longer feasible for economic or other reasons. After the final shutdown of the NPP, all fuel from the core and radioactive fluids from the reactor systems are removed. The NPP is then kept in a preserved state with appropriate surveillance till the dismantling of its structures and components is taken up for its total decommissioning. This waiting period can be several years or even a few decades in duration and is decided by factors like the need for allowing natural decay of radioactivity in the reactor structure and components to bring down radiation fields before starting the dismantling. Another factor could be the requirement of making the site available for setting up new NPPs or for its unrestricted release for public use or for other purposes.

It is necessary to develop the required expertise in advance for dealing with various aspects of decommissioning. During the design safety review prior to the start of construction, a thorough check is made to ensure that necessary provisions and features are incorporated in the design to the extent feasible to facilitate decommissioning at the end of the design operating life of the NPP. These would include aspects such as the use of materials in the neutron flux region that do not generate activation products with very long half-lives, careful segregation of components that will become radioactive during operation from those that will not get activated with the aim of reducing radioactive waste volumes from decommissioning, and provision of material handling facilities for removal of highly radioactive components with minimum possible radiation exposure of plant personnel. During operation of the NPP care has to be taken to prevent spread of radioactive contamination that would unnecessarily increase active waste quantities during decommissioning. Also, records must be meticulously maintained of any spread of contamination if it occurs and of plant modifications and introduction of new materials that might generate radioactivity, especially with long half-life radionuclides.

During dismantling for final decommissioning, large quantities of radioactive waste of various types will be generated and its disposal would pose a significant challenge. Technical expertise will need to be developed for the handling, volume reduction, packaging, transportation and disposal of such waste. Criteria will also have to be developed for releasing materials from decommissioning for reuse in a nuclear facility or for unrestricted use. Dismantling of highly radioactive components is another challenging task during decommissioning. Appropriate technologies and tools have to be developed to be able to carry out remote dismantling where necessary to minimize radiation exposure of personnel. In some cases dismantling may have to be done under water to minimize exposure of personnel and to prevent generation of airborne radioactivity. Experience from maintenance work carried out during NPP operation will be useful in some cases for such development work and therefore it needs to be recorded properly. The techniques, tools and procedures for complex jobs should be developed and qualified well in advance of actual decommissioning and the technical support organizations will have to play a major role in these activities.

7.10 Sources of further information and advice

Food and Agriculture Organization of the United Nations, International Atomic Energy Agency, International Labour Organization, Nuclear Energy Agency of the OECD, Pan American Health Organization, World Health Organization, *International Basic Safety Standards for Protection against Ionizing Radiation and for the Safety of Radiation Sources*, Safety Series 115, IAEA, Vienna (1996)

International Atomic Energy Agency, *Near Surface Disposal of Radioactive Waste, Safety Requirements*, IAEA Safety Standards Series No. WS-R-5, IAEA, Vienna (1999)

International Atomic Energy Agency, *Safety of Nuclear Power Plants: Design Safety Requirements*, IAEA Safety Standards Series No. NS-R-1, IAEA, Vienna (2000)

International Atomic Energy Agency, *Safety of Nuclear Power Plants: Operation Safety Requirements*, IAEA Safety Standards Series No. NS-R-2, IAEA, Vienna (2000)

International Atomic Energy Agency, *Site Evaluation for Nuclear Installations*, IAEA Safety Standards Series No. NS-R-3, IAEA, Vienna (2003). Detailed guidance in fulfilling the site requirements is provided in the related IAEA Safety Guides NS-G-3.1 to NS-G-3.6

International Atomic Energy Agency, TECDOC 1513 *Basic Infrastructure for a Nuclear Power Plant*, IAEA, Vienna (2006)

International Atomic Energy Agency, *The Management System for Facilities and Activities*, IAEA Safety Standards Series No. GS-R-3, IAEA, Vienna (2006)

International Atomic Energy Agency, TECDOC 1555 *Managing the First Nuclear Power Plant*, IAEA, Vienna (2007)

International Atomic Energy Agency, *Arrangements for Preparedness for a Nuclear or Radiological Emergency*, IAEA Safety Standards Series No. GS-G-2.1, IAEA, Vienna (2007)

International Atomic Energy Agency, *Milestones in the Development of a National Infrastructure for Nuclear Power*, IAEA Nuclear Energy Series No. NG-G-3.1, IAEA, Vienna (2007)

International Nuclear Safety Group, *Strengthening the Global Nuclear Safety Regime*, INSAG-21, IAEA, Vienna (2006)

International Nuclear Safety Group, *Nuclear Safety Infrastructure for a National Nuclear Power Programme Supported by the IAEA Fundamental Safety Principles*, INSAG-22, IAEA, Vienna (2006)

International Nuclear Safety Group, *Improving the International System for Operating Experience Feedback*, INSAG-23, IAEA, Vienna (2008)

7.11 Acknowledgements

The author wishes to thank Mr S. S. Bajaj, Chairman of the Atomic Energy Regulatory Board of India, who painstakingly reviewed the manuscript of this chapter and made important suggestions for its improvement. Thanks are also due to Mr M. K. Damodaran, my former colleague in the Atomic Energy Regulatory Board, who helped with the typing of the manuscript of this chapter in his spare time.

8 Application of the justification principle to nuclear power development

A. ALONSO, Universidad Politécnica de Madrid, Spain

Abstract: This chapter outlines the justification principle and how it could be applied to taking decisions about the development of nuclear power. The justification principle compares the economic, social and environmental benefits derived from a given development against the risks and detriments associated with it. When the benefits outweigh the associated risks and detriments, the intended development is considered to be justified. The benefits from using nuclear power for the generation of electricity arise from its reliability, independence, costs and freedom from carbon emissions, while its risks and detriments are associated with the generation of toxic radioactive products and strategic nuclear materials that need to be kept under control. The justification principle could be used to consider the establishment or continuation of a nuclear development plan, to select an individual design and the corresponding fuel cycle, or to help decide the longer-term operation of an already operating plant.

Key words: benefits of nuclear power, detriments from nuclear power, ethics of justification, justification equation, justification process.

8.1 Introduction

The justification of facilities and activities is the fourth principle of the 10 'Fundamental Safety Principles' introduced by the International Atomic Energy Agency (IAEA) in 2006. The principle states that 'facilities and activities that give rise to radiation risks must yield an overall benefit' (IAEA, 2006). This principle was first introduced by the International Commission on Radiological Protection (ICRP) as a basic principle in the protection against ionizing radiation (ICRP, 1990) and has mainly been applied in medical and other uses of radiation.

Many countries have introduced the justification principle into national legislation, although limiting its application to radiation protection in radiation uses. The definition of justification given by the IAEA extends the application of the principle to facilities and activities where radiation risks are present; nuclear power plants and related fuel cycle installations and activities lie within this class. Although such installations and activities are clear candidates for application of the principles, many countries have not included such requirements in their general regulatory practices, with the notable exception of the United Kingdom for general applications (UK, 2004) and specifically for nuclear power designs (UK, 2008), as explained in Appendix 1.

Supporters of the application of the principle to the development of nuclear power believe that it serves to balance the benefits and detriments, providing insights into high-level decision processes and aiding the social acceptance of nuclear energy. When solving the justification equation, all possible benefits should be assessed as well as all risks and detriments coming from the construction and operation of nuclear power plants, fuel cycle installations and related activities. Benefits, risks and detriments can be economic, social or environmental. All these elements will be described in this chapter.

The International Nuclear Safety Group (INSAG) advises that the 10 IAEA Fundamental Safety Principles be applied to all the different phases in the life of a nuclear power plant (INSAG, 2008). The application of justification to all phases in the life of a nuclear power plant could be very effective for the early phases, and it should be considered as part of the decision to launch a nuclear power programme and in the selection of acceptable technologies. During plant operation, it could also help when taking decisions such as enlarging the capacity and long-term operation of a nuclear power plant, as well as assisting in extraordinary circumstances, such as when recovering from relevant incidents, and equally could also be used in selecting the decommissioning level and the technology used for it.

INSAG recommends that the justification principle should be applied by new entrants and by those countries interested in expanding their nuclear power programmes (INSAG, 2008). Nevertheless, neither INSAG nor the IAEA have developed detailed technical guidance on how to develop a justification document. The purpose of this chapter is to give such guidance on how to develop the terms included in the justification equation.

The application of the justification principle needs a process and a justification authority. A country's government is responsible for establishing the regulatory requirements and corresponding guidance, and for selecting the justification authority. Relevant decisions, such as the decision by a state to embark on a nuclear power programme, are generally taken at the highest levels of government. For other decisions, such as for those concerning the longer-term operation of existing nuclear power plants, the regulatory body may determine whether the decision is justified.

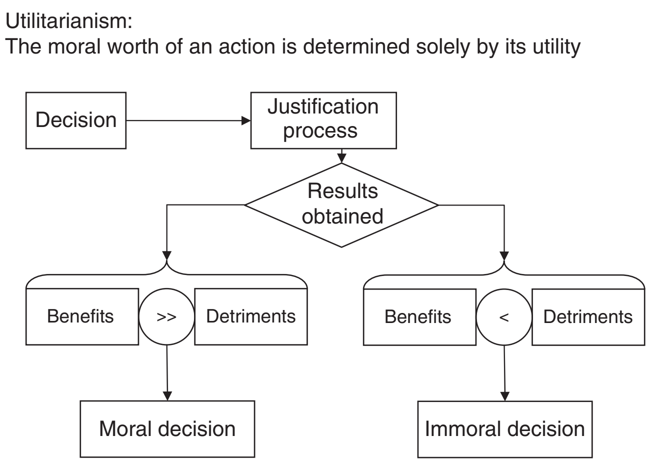
8.2 The ethics of the justification principle

The broader meaning given in the IAEA Fundamental Safety Principles (IAEA, 2006) to the original ICRP-defined justification principle, and particularly its application to justification of nuclear energy development, leads easily to the conclusion that such an application is based on ancient so-called teleological ethics which can be expressed in the sentence 'the ends justify the means'. The application of this ethical principle was the basis of the utilitarian ethics developed by the British philosopher Jeremy Bentham (1748–1832), which considered things moral and therefore acceptable if they achieved 'the greatest good for the greatest number of people'.

Utilitarians hold that an action is moral when the good consequences of an action outweigh the bad. Utilitarianism was further developed by John Stuart Mill (1806–1873), another British thinker. Utilitarian ideas are alive in the justification principle in the sense that the benefits of any decision should be a determining factor in judging its morality. The decision process within utilitarian ethics is presented in Fig. 8.1. Any potential action should be analysed to determine its benefits and detriments: if the former dominate the latter, the action is moral and it should be taken; if the opposite is true, taking the action would be immoral.

Despite its global acceptance, utilitarianism has some inherent difficulties:

- Achieving an acceptable end does not justify the means: the benefits from nuclear power may be well recognized but the means to achieve them are still not acceptable to many.
- There is no equity in the distribution of benefits and detriments: the benefits of nuclear energy may affect the whole world, as in the case of carbon abatement, or an individual country, in the case of having security of service and lower electricity prices, but most of the inconveniences go to the immediate neighbouring population.



8.1 Decision process in utilitarian ethics. A decision is moral, therefore acceptable, when benefits surpass detriments.

- It is not always possible to predict the benefits and the detriments with certainty, as in the case of avoiding the proliferation of nuclear weapons and the expected frequency of potentially catastrophic accidents.
- The results of a justification analysis (quantifying benefits and detriments) should also be judged when they are compared. Benefits and detriments from nuclear power cannot always be measured with the same metrics: it is possible to assess risk quantitatively but it is not possible to quantify perceived risk.

The means to obtain a benefit constitute a major ethical issue and a difficulty in the acceptance of the justification analysis. In most countries, the development of a nuclear energy programme is mainly a political decision; such a programme includes provisions for the construction of nuclear units, the selection of a nuclear fuel cycle, the management of used fuel, and the management of radioactive wastes. When the justification process is undertaken rationally, by independent groups of experts using the best tools available, and considering all the benefits, risks and detriments on their own merits, the most probable outcome is a decision to develop the programme, because the benefits outweigh the detriments.

Such an exercise should be transparent and formally presented for public consideration. When such an exercise is analysed by other institutions, those dominated by radiation fear and with an exaggerated perception of the risks will conclude that the means used, i.e. nuclear reactors and fuel cycle facilities and related activities, do not justify the decision and will stick to the use of other sources of electricity. This problem can be addressed by closely analysing the safety of the nuclear plants and fuel cycle facilities to be built, and the activities to be conducted there. Proving that such installations and activities will be constructed and operated in accordance with the IAEA Fundamental Safety Principles and related standards should guarantee an acceptance of the means used in the justification process.

The intrinsic lack of equity in the distribution of benefits and detriments is another impediment in the development of nuclear energy. Utilitarianism does not protect the rights of everybody equally, although the benefits should go to the greatest number; the benefit of enjoying lower electricity prices is a national and collective asset, but the individual benefit is proportional to the amount of electricity each person consumes. By contrast, radiological risks and potential detriments are larger for those closer to the installations. This issue is addressed by compensating detriments with taxation and subsidies, in addition to the rather positive direct and indirect advantages derived from the installations within local, county, provincial and state territories.

Uncertainties about the basic data and the tools available to determine benefits and detriments, and difficulties in using the same metrics, are also serious impediments in any final comparison exercise. Some of the tools available, mainly those used for economic analysis and risk estimation, are well developed but the data used may be uncertain. To cope with those problems, analysts make projections by varying key parameters. Ensuring the use of the same metrics is more complicated: to allow it, a so-called alpha value has been developed to assign a monetary value to the radiation dose received or averted, but this technique is not globally accepted.

The results of any decisions taken should be analysed to determine if any predictions made have been achieved, and to introduce corrections if necessary. Intermittent natural energies, mainly wind and solar, are being promoted by supranational organizations, such as the European Union, and by individual countries, while other countries have developed ambitious nuclear development programmes; the impact of the Fukushima event will cancel or retard the execution of some of these programmes while others will continue as originally established. There will be countries where nuclear energy will be promoted, while in others nuclear energy may be banned completely, and there could also be countries using a diversity of energy sources. In the coming decades, each country will have an opportunity to check the soundness of the decisions they have taken, and in each case the public has the right to be informed.

8.3 The justification process

Any justification exercise, whether relating to a simple new medical or radiation application, a new nuclear power programme or any other nuclear installation or relevant activity, should be undertaken within a formal regulatory process. Such a process will provide a list of projects which can be submitted for justification, define the corresponding justification authority, incorporate guides for submitting the different types of justification, provide for effective stakeholder participation and define the legal value and scope of any decision taken.

The justification principle is included in most regulatory requirements concerning radiation protection but has only been developed for radiation applications in some countries, and only in the United Kingdom have regulations been fully developed for justification of nuclear power installations. In 2004, new regulations were enacted to develop the justification principle (UK, 2004). These regulations included 27 articles and four schedules; among other legal aspects, they define those installations and practices that should undergo a justification request, the process for requesting justification, the authorization authority and public participation in the process.

In March 2008, the UK Secretary of State for the then Department for Business, Enterprise and Regulatory Reform (BERR), in consultation with the Department for Environment, Food and Rural Affairs (DEFRA), issued a guidance document (UK, 2008) specifically aimed at applicants wishing to seek a decision under the justification regulations to justify new nuclear power. That guidance sets out the process for submitting applications and outlines the decision-making process for such justification. The guidance note defines justification as 'a high-level assessment to assess the benefits and any health detriment associated with a particular class or type of nuclear practice'. It clearly indicates that 'it does not, by itself, authorise the construction or operation of any particular plant or activity, nor does it replace the detailed safety, security and environmental assessments carried out by the nuclear regulators'.

The BERR interpretation of justification requires an assessment of the potential radiological health detriments associated with the practice, but also other potential detriments that could be significant when considered against the benefit derived from the practice. Following this guidance, the Nuclear Industry Association (NIA) submitted to the Department of Energy and Climate Change (DECC) – considered to be the justification authority in this particular case – an application for justification of new nuclear power stations (NIA, 2008), with reference to the two designs (the AP1000 from WEC, and the EPR from AREVA) which, at that time, had successfully completed Step 2 of the Generic Design Assessment (GDA). This subject is developed further in Appendix 1.

The responsibility for preparing, drafting and submitting an application for justification may lie with an *ad hoc* group of specialists, with any industry or association of industries, or with any licensee responsible for a given installation. Subjects to be justified may include a national nuclear power programme, a cluster of nuclear power designs, a nuclear research centre (including research reactors), a fuel cycle management policy, or the transportation of nuclear materials and radioactive waste.

In all cases, there should be a justification authority. Indeed, the nature and aims of the justification principle require the existence of a justification authority with the capacity to decide whether a given request should be considered justified. The rank of such an authority may vary in accordance with the magnitude and the nature of the issue to be justified. The decision to introduce a nuclear power programme for the first time, or to construct large installations, should lie under the authority of the head of government or a suitable minister of state, generally under parliamentary control. The decision to build a final repository for long-life high-level waste should also be the decision of the government. In both cases, the basis for the decision should be a justification exercise produced by national experts and incorporating external advisers, when needed.

The decision to enlarge the service life of operating nuclear power plants beyond the life assigned in the original design should be taken by a minister of state or similar authority responsible for energy. A justification report should be prepared and submitted by the plant owner/operator to the justification authority. In this particular case, the nuclear regulatory organization has to verify that such longer-term operation will maintain the design basis, and that the ageing process can be managed. The justification will not impede or impair the safety evaluation of individual plants by the regulatory authority; it simply ensures that such types of request are considered properly.

The process should be open to stakeholder involvement through a formal process, as explained by the IAEA International Nuclear Safety Group (INSAG) (INSAG, 2006). The intensity and coverage of stakeholder participation may vary considerably: it may cover the whole country, and even neighbouring countries, such as when considering embarking on a new nuclear power programme or deciding the location of a final waste repository. It could be limited to a particular state or province and their neighbours, as in the case of the construction of a new nuclear installation, or be limited to the neighbourhood and the area of influence of an existing installation. But whatever the case, there should be a well-established procedure for stakeholder intervention, and the justification authority should make its decision after a careful analysis of all their submissions.

A justification decision only implies that the issue requested is acceptable and can be put into practice, for example that a proposed nuclear power programme can be conducted within the limits and conditions stated in the decision, and that nuclear installations and relevant activities are acceptable as described. The decision is not a licence and does not compromise any regulatory decision, nor the regulatory process itself. One of its main values resides in the fact that the decision takes into account other types of consideration beyond nuclear safety and security, and radiological protection, and that it has considered the opinion of society at large.

8.4 The terms of the justification equation

Justification has to prove that the benefits from a programme, or from the installation of the activity analysed, will override the ensuing risks and detriments. Therefore, all terms in the equation have to be defined and quantified to the best possible level. Not all elements can be quantified, nor do they use the same metrics. Moreover, not all benefits, or the risks and detriments, relate to the same recipients. Economic, social and environmental benefits should apply to well-defined receptors, defined as follows:

- *The world*: a reduction of the geopolitical tensions created by the depletion of conventional fuels; a reduction of carbon emissions; the enhancement of worldwide commercial activities and technology interchanges

- *The nation*: an improvement of the country's energy independence; an upgrading of the reliability and security of electricity production; an improvement in the country's scientific and technical expertise
- *Individuals within the area of influence*: an increase in monetary revenues through taxes and subsidies; a development of business and commercial transactions; a reduction of unemployment.

Similarly, risks and detriments also have an effect on the same receptors:

- *The world*: expansion of proliferation risks; an increase of radiation risks coming from worldwide activities related to fuel cycle activities; a growth in the international transportation of nuclear materials and radioactive waste
- *The nation*: an increase in the final repository of radioactive waste; an increase in activities related to emergency management; the radiological environmental impact of installations and related activities
- *Individuals within the area of influence*: risks from radiation exposure to radioactive effluents (planned exposures); risks associated with emergency situations (potential exposures); non-radiological environmental impacts.

Some of the items above are amenable to quantification in monetary terms or by other means, but most of them are subjective and country-dependent. The items amenable to quantification will be considered in detail, whilst those which are subjective are treated as such in the following paragraphs. Both the benefits and the risks and detriments are closely associated with characteristics specific to nuclear energy, and discussion of them constitutes the backbone of this chapter. The risks and detriments come from the need to prevent and mitigate accidents with radiological effects, the generation of radionuclides by fission and activation, and the generation of strategic materials.

8.5 The benefits of nuclear energy

The benefits of nuclear energy are related to its potential to replace fossil fuels, reduce carbon emissions and hence control climate change, and the potential increase of scientific and industrial progress deriving from an intellectually and technologically intense activity, together with the social and economic development of the societies affected.

As discussed in the previous section, the whole world benefits from the substitution of oil and gas by nuclear power for the generation of electricity, as a result of the corresponding decrease in the emission of greenhouse gases, and by the increase in commercial activities and technology interchanges between nations. Similarly, the whole of a country benefits from the increased reliability and potential lower costs of electricity, and local and regional populations benefit from taxes and subsidies, and from the direct and indirect economic and developmental effects of nuclear activities. All these benefits are closely related to particular characteristics of nuclear power, which will now be considered in the following sections.

8.5.1 Nuclear power is capital intensive

The costs of electricity generated by different sources should include the cost of the plant, the cost of the fuel, and the cost of operation and maintenance (the economics of nuclear power are considered in detail in Chapter 15 of this book). The cost of the nuclear plant is the most relevant of the three component costs which, all considered, make nuclear power the cheapest producer of base load electricity (though only if discount rates are reasonable and the plant can be put into operation as designed). Long delays caused by licensing requirements, equipment supplies or other causes can change that situation, this being the reason why utilities insist on reliable licensing processes and government guarantees.

Prices (quoted here in US dollars per kilowatt of electric power) vary considerably. The cost of plants built recently in Japan and South Korea has been quoted as close to $3000 per kW, while the Olkiluoto and Flamanville plants under construction in Finland and France, respectively, may cost more because of delays in construction.

Many national and international institutions constantly estimate the costs of electricity from various sources. The NEA/OECD, in cooperation with the International Energy Agency, estimates costs on a regular basis and provides updates (OECD, 2011). Likewise, industry institutions such as the World Nuclear Association (WNA) also provide updates on nuclear power plant economics (WNA, 2011). In all cases, electricity costs from nuclear power are comparable with those from coal, and are cheaper than those for gas and renewable sources. When a carbon tax is imposed on coal and gas, nuclear power becomes the most competitive source.

8.5.2 Nuclear fuels as a substitute for fossil fuels

Fossil fuel reserves and potential resources are under constant evaluation. New deposits are found by exploration and by applying new extraction technologies, but the consumption rates of such fuels are increasing, mainly in developing countries. Because such resources are finite, they cannot be sustainable for a long time. Coal deposits are more abundant that oil and gas, but they too will come to an end. To avoid geopolitical tensions, it is necessary to use new sources of electricity production; it was this need that was at the root of the development of nuclear power for electricity production. The world growth of electricity production and the depletion of fossil fuels are the reasons why international institutions (NEA, 2008) are encouraging the construction of new nuclear power plants, and why individual countries, even countries with large reserves of gas and oil (such as the United Arab Emirates), have already embarked on the construction of nuclear power plants.

The steady substitution of oil and gas by nuclear power stations will moderate the effects of the increasing unavailability and potentially increasing prices of oil and gas as reserves diminish. Although uranium and thorium resources are large, they are also finite and will only be made sustainable for many centuries with the introduction of fuel reprocessing and breeder reactors. Such technology is already available. Fuel reprocessing is commercially conducted in France, the UK and other countries; fast breeder reactors, up to a technological and even commercial demonstration level, have been operated for years in France, the UK, Russia and Japan. New developments are now being considered and there are no intrinsic problems that could prevent the full commercial deployment of such technologies. Although these considerations are difficult to evaluate in numerical terms, they are clearly on the benefits side.

8.5.3 Reduction of carbon emissions

Scientific interest in understanding climate changes due to greenhouse gases began in the late nineteenth century with the Swedish chemist Svante Arrhenius, and such interest has developed considerably since that time. The history of such developments is described by the science historian Spencer R. Weart in his book *The Discovery of Global Warming* (Weart, 2008). Today there is no doubt that global warming, first noticed during the industrial revolution, is due to the increase of greenhouse gases, mainly CO_2 from anthropogenic sources, of which electricity generation is a part. The effort now is being put into determining the relationship between CO_2 in the atmosphere and the increase in air temperature at soil level.

The United Nations sponsored Intergovernmental Panel on Climate Change (IPCC), created in 1987, is currently the international organization responsible for reviewing and consolidating research on climate change and its effects. Since all nations share the atmosphere, climate change affects everyone, so control of carbon emissions is a global bonus. Any efforts made to reduce such emissions by using nuclear power instead of fossil fuels is a benefit for the whole world.

The IPCC produced major assessments on the climate change situation in 1990, 1995, 2001 and 2007. The first report underscored the seriousness of the risks associated with climate change and it was the driver for the 1992 UN Earth Summit in Rio de Janeiro, which led to the UN Framework

Convention on Climate Change (UNFCCC) and, later, in 1997, to the Kyoto Protocol. A clash between the requirements of developing and developed countries delayed the entering into force of the Protocol to 2005 and limited its validity to 2012. The Protocol established that developed countries should achieve an average 5.2% cut in CO_2 emissions by 2008–2012, when compared to 1990 levels (UN, 1998). The Protocol created an emissions market and defined the so-called Clean Development Mechanism (CDM) which is non-applicable to nuclear projects. Developments expected after 2012 are not yet well defined.

One of the major worldwide advantages of nuclear power is its limited greenhouse gas emissions and its therefore corresponding contribution to a reduction in climate change (UIC, 2001). The nuclear fission reaction is anaerobic, i.e. it does not need air to generate energy, as is the case with fossil fuels. Fossil energy may be needed in the nuclear fuel cycle, for uranium mining and milling, conversion, enrichment, fuel fabrication, reprocessing and waste deposition and related transportation activities. The fabrication of components, construction and assembly of a nuclear power plant and its dismantling need fossil energy in the same way as other electricity generating installations of a comparable size. The operation of a nuclear power plant is, however, generally free from carbon emissions, except for some safety and ancillary equipment such as emergency diesel generators, which have to be tested periodically, and boilers for heating sanitary and process water.

The release of CO_2 from the different sources of generating electricity has received considerable attention. Up to the year 2000, it was estimated that nuclear energy could release up to 16 t CO_2/GWh (Spadaro, 2000), while the release from coal and natural gas could amount to 1100 and 450 t CO_2/GWh, respectively. These data are approximations that have been recently refined. First, there are differences in the type of coal and the thermal efficiency of the plant being considered: lignite can produce 1200 t CO_2/GWh, while hard coal is limited to 1070 t CO_2/GWh and can even go down to 974 t CO_2/GWh for modern high-efficiency plants. Figures for gas combustion in conventional stations can be 650 t CO_2/GWh, down to 450 t CO_2/GWh for modern combined cycle plants. There are also variations in nuclear power plants mainly due to the enrichment process used: the gas diffusion process needs close to 50 times the energy needed in the gas ultracentrifugation process, and it can be as low as 5 t CO_2/GWh. The data quoted here are taken from a number of different sources (NEA, 2008; Richter, 2010).

The data quoted for nuclear power include so-called plant life-cycle emissions, made up of the CO_2 released in making the steel and concrete used in the plant, as well as that generated during dismantling and radioactive waste management, divided by the energy produced by the plant during its expected lifetime; to this is added the emissions involved in fuel cycle activities, including transportation and the limited direct emissions from operation. Longer-term operation of nuclear power plants therefore reduces their carbon footprint. This concept also applies to other carbon-free power sources, such as wind and solar power; as for nuclear power, the CO_2 footprint during operation of these sources is limited to maintenance and ancillary operations. Within this context, CO_2 emissions from wind turbines are comparable to those of nuclear power, while those of solar power are two to three times larger.

With the basic data provided above, it is possible to determine the CO_2 emissions that are avoided by using nuclear power instead of coal or gas. For one GWe nuclear plant operating with a 90% capacity factor, 7.6–9.3 million tons and 3.5–6.2 million tons of CO_2 are effectively saved per year compared to that generated if the same energy were produced by coal and gas plants, respectively (depending on the technology used and the type of coal). The CO_2 avoided carries a monetary value when using the Cap and Trade scheme already practised within European Union member states.

A recent report produced by the United Kingdom Committee on Climate Change states that 'nuclear generation in particular appears likely to be the most cost-effective form of low-carbon power generation in the 2020s (i.e. before costs of other technologies have fallen), justifying significant investment if safety concerns can be addressed' (CCC, 2011). Similar results have been found by the International Energy Agency in its economic evaluation of ways to reduce the carbon content of the atmosphere to 500 ppm by 2050.

8.5.4 Worldwide commercial activities and technology interchanges

Nuclear power has a distinct global dimension and its potentially widespread deployment will bring to the world an intense commercial and fruitful technological interchange, with the potential to improve other technologies too. It demands modern science and high technology and requires a complex fuel cycle and, as such, its global introduction will create an exchange of experts who will disseminate scientific and technological knowledge and experience for the benefit of every country involved.

During the pioneering years, the so-called nuclear countries developed many different technologies for the peaceful use of nuclear power. Although many prototypes were tested, today those technologies have been reduced to light water reactors (LWR) in the form of pressurized water reactors (PWR) and boiling water reactors (BWR), first developed in the United States and in the old Soviet Union, and heavy water reactors (HWR), the CANDU models, which have been developed in Canada and India. The UK chose to continue with their advanced gas-cooled reactors (AGR). Other industrialized countries have developed their own copies: France, in particular, developed its own PWR models, the former West Germany several PWR and BWR versions, and Sweden a BWR reactor system. France and Germany also developed the EPR model which is now promoted by the French company, Areva.

Other countries, in particular Japan, South Korea, Italy and Spain, bought several PWR and BWR models and established a well-developed scientific and technical infrastructure. In 1987, Italy decided to cancel and dismantle all its nuclear power plants, whilst in 1983 Spain decided to establish a moratorium on the construction of new nuclear power plants. By contrast, Japan and South Korea decided to continue their nuclear development and have now become providers of nuclear designs. The now united Germany decided in 2000 to dismantle its well-developed nuclear industry. More recently, after installing different foreign models, China has been able to develop its own PWR model.

It is expected that light water reactors (with possibly a few heavy water reactors) will be the preferred option in the near future, supplied by a limited number of providers. The country importing the technology will have an opportunity to participate in the design, manufacturing of components, assembly and construction of its plants, and will be responsible for their operation and the management of radioactive waste and used fuel. Moreover, the technology associated with the fuel cycle is equally complicated and global. Uranium mining and milling could be performed by nationals of the countries where reserves are found. Enrichment and fuel manufacture are more complex technologies but they can be managed in many countries. Reprocessing is more technology intensive and non-proliferation sensitive, and may not be open to all. The activities mentioned above need international transportation of heavy components, radioactive materials and nuclear fuels. All these activities create positive international commerce and a transfer of technology.

8.5.5 Energy independence and security of supply

A nation developing nuclear power or simply building a new power station benefits from improved energy independence from the outside world, with upgrading of the reliability and security of electrical production, and promotion of the educational, scientific and technical development of the country.

Countries appreciate being energy independent from other countries. Energy independence means supply security and price stability. Countries which are dependent on energy from external suppliers cannot control their economies and can be the subject of energy embargoes – circumstances which have occurred frequently, historically. The energy policies of most countries strive to maintain their independence from other countries as much as possible. Energy independence should not be confused with interdependence, i.e. mutual dependence, which favours trade and interchange among the parties.

Energy independence is achieved by substituting fossil fuels with domestic products, such as developing nuclear power that directly replaces fossil fuels for the generation of electricity. In fact, energy independence is a major driver for nuclear power. The UK government was not interested in new nuclear builds until it realized that its gas and oil reserves in the North Sea had been exhausted and that it had to import such commodities. Many European countries are heavily dependent for energy on Russian gas and on oil from the Gulf and North African countries, and their energy situation is vulnerable. Because of nuclear phobia (very intense in some central European countries) and the risks associated with climate change produced by CO_2 emissions, the current policy of the European Union is to develop wind and solar power, probably beyond their technical and economic possibilities. Such developments are only possible if they are heavily subsidized, which has a negative impact on economic development.

Replacing fossil fuel by nuclear energy does not necessarily make countries completely energy independent, but certainly improves the situation. The approximate specific energy delivered by natural gas is 55 MJ/kg and half that amount for hard coal, while low enriched uranium in current LWRs can produce some 3.9×10^6 MJ/kg. This considerably simplifies fuel transportation and storage issues. Moreover, uranium resources are more evenly distributed than gas and oil; reserves are abundant and the volatility of prices more limited. Apart from that, the cost of the fuel cycle represents only some 15% of the generated electricity cost, from which only 5% corresponds to the price of the natural uranium.

To assess the benefits obtained from gaining energy independence by selecting nuclear power, the volatility of fossil fuel prices and the stability of nuclear fuel pricing have to be compared, as well as the cost of storing such fuels to control supplies and the impact of fuel on the production cost of the electricity generated.

The 15 millennia of accumulated operating experience of the world's nuclear power plants has proved that, on average, they can now operate reliably within load factors close to 90%. Some plants have refuelling outages every one to two years, lasting three to six weeks, and generally operate continuously at nominal power in between outages. Although they can change power, these plants are designed to provide load-based electricity and are not suitable for following demand. In this sense, they cannot provide backup power for intermittent sources, such as solar and wind, but can be good substitutes for large coal and gas power stations.

The net benefits provided by this characteristic are country dependent. A nuclear power plant which is part of a national electricity grid can provide stability to the grid whilst, for safety and operational reasons, the nuclear plant itself requires the grid to be stable at the same time. It is essential that there is an equilibrium between generation and demand; when this equilibrium is lost, the grid becomes unstable and there could be limited or even complete blackouts. To avoid such situations, generating plants have to be able to provide primary regulation (within seconds) for small fluctuations, secondary regulation (within minutes) for larger perturbations, and also tertiary regulation (within hours) to fully recuperate any perturbed equilibrium. Nuclear power plants have the capability of reliably maintaining power and responding to small fluctuations but they are not normally used for secondary regulation.

8.5.6 Impact on educational, scientific and technical development

Nuclear science and technology is highly demanding intellectually, and nuclear deployment requires a high level of expertise in human resources (as presented in Chapter 6 of this book). Past experience has shown that the introduction of nuclear power in a country can be the driver for the establishment of new educational programmes, scientific and technological institutions, and organizations and research centres. As nuclear science and technology also have other uses, these new institutions and activities can be considered beneficial for a country as a whole.

The stagnation of nuclear development created in many countries after the Three Mile Island unit 2 (TMI-2) and Chernobyl-4 accidents was immediately detected in educational systems. Nuclear courses that were very prominent and well attended in the 1970s and during the first half of the 1980s in European and American universities almost completely disappeared. Most of the high-level experts prominent in those years are now entering retirement age, and thus a gap in high-level human resources is growing. This situation has been recognized by international and supranational organizations such as the IAEA, the NEA/OECD and the European Council, as well as by leading nuclear technology countries. INSAG has also voiced its concern over the need for human resources in nuclear safety research (INSAG, 2003) and new educational programmes have been created to cope with the situation.

The IAEA has created a new series of teaching modules and materials which are described in Appendix 3 of this book. A World Nuclear University (WNU) was created within the World Nuclear Association (WNA) and the World Association of Nuclear Operators, which also has the support of the

IAEA and the NEA/OECD, and which includes leading universities and nuclear education institutions in more than 30 countries. The WNU is a 'global partnership committed to enhancing international education and leadership in peaceful applications of nuclear science and technology'.

Similarly, in Europe, a programme (within the fifth framework programme for research and training activities) was launched on high-level nuclear engineering education, giving rise to the European Nuclear Education Network (ENEN), a non-profit association formed in 2003. As of March 2011, the ENEN has 60 members and partners in 18 EU countries, South Africa, the Russian Federation, Ukraine and Japan, consisting of 33 effective members, primarily academics, and 27 associate members, including nuclear research centres, industries and regulatory bodies.

A similar organization, the Asian Network for Education in Nuclear Technology (ANENT), was created within the auspices of the IAEA to serve the Asian countries, 'to promote, manage and preserve nuclear knowledge and to ensure the continued availability of talented and qualified human resources in the nuclear field in the Asian region and to enhance the quality of the resources for sustainability of nuclear technology'. As of May 2011, the ANENT network had 17 State Members, six Collaborating Members and six potential Collaborating Members. In a similar way, many countries are fostering high-level nuclear education with positive results, and education and training at the technician level has also been fostered in many countries and organizations.

Research and development also declined during the stagnant period following the TMI-2 and Chernobyl-4 accidents, with the exception of research into severe accidents, nuclear safety research into operating nuclear power plants, and the management of radioactive waste and used fuel. Research on severe accidents was increased in the USA after the TMI-2 accident within an international context initiated by the International LOFT Project. Research projects were undertaken on all associated phenomena, including an investigation of the behaviour of the molten core when outside the pressure vessel. The knowledge gained has been used to improve the design of new reactors and presented at many national and international conferences, as part of the Euratom-driven FISA meetings (FISA, 2001, 2003, 2006).

Nuclear safety research into how to operate nuclear power plants is necessary to understand the ageing mechanism and to provide information for the longer-term operation of these plants. The NEA is the international organization of reference, publishing documents on research needs. INSAG has also expressed concerns about the importance of nuclear safety research (INSAG, 2003).

Countries with already operating nuclear power plants as well as new entrants building their first nuclear power plants should boost education at university and vocational level, and reinforce or create nuclear research centres, participating in international research projects commensurate with their needs. There will be direct and indirect benefits as a result of such efforts. Once a nuclear power plant is transferred from the reactor supplier to a national operating organization, the primary responsibility for its operation rests within the licensee, under the supervision of the regulatory authority. That responsibility requires knowledge and expertise and cannot easily be transferred to contractors.

Knowledge and expertise of nuclear matters also gives indirect benefits such as an improvement of the scientific and technical development of the country, which can be applied to other industries and activities. An evaluation of these benefits can be made by analysing the technical and scientific developments which other former entrant countries have achieved.

8.5.7 Taxes and subsidies developing the area of influence

Enquiries conducted among people living near operating power plants show a positive acceptance of nuclear energy. There are at least three reasons for that acceptance: the socio-economic benefits from the nuclear power plant; growing confidence in the operators through a policy of transparent dialogue and information; and the remote perception of a nuclear risk. In this section, the socio-economic benefits of a nuclear power plant are examined.

Most nuclear power plant owners have conducted studies on the socio-economic impacts that they produce in their areas of influence. The US Nuclear Energy Institute (NEI) has so far conducted 13 such studies which include 22 nuclear units, from which some general statements have been published. Other institutions, mainly university departments, have conducted analyses for other plants (Exelon, 2008). Moreover, local economic impact assessments have also been conducted for decommissioning (PG&E, 2010) and for the Yucca Mountain nuclear waste repository (University of Nevada, 2003).

The methodology used is based on so-called input–output analysis, available on the market. The application of such models is explained in Section 6 of the NEI analysis of the economic benefits that the institution has so far conducted (NEI, 2008). A number of commercial models are available, with Impact Analysis for Planning (IMPLAN) being the one used in the NEI evaluations.

IMPLAN analyses the interrelations between input-demand and output-supply for any activity, such as the construction, operation and dismantling of a nuclear power plant, within a defined geographical region. The aim of the analysis is to determine the expenditures that the plant will bring to the region, the income generated for local businesses and households, the number of jobs that the plant may provide for the different stages in its life, and the tax revenues generated.

The impacts of the plant on the regional area of interest will include not only direct impacts, i.e. the initial impacts from bringing the plant to the region, but also secondary effects produced by those first impacts, i.e. the demand for goods and services will itself generate new employment and additional spending to deliver the goods and services requested by the plant and by other potential customers. The addition of the two effects is called the total effect, and the ratio between total and direct effects is called the multiplier effect, which can be obtained for each individual effect, such as an increase in jobs, earned income, industry output or revenue from taxation. Multipliers can be obtained for local, county, provincial and state areas.

From the experience obtained through these studies, NEI has published a Fact Sheet which summarizes the contributions of nuclear power plants to state and local economies in the US (NEI, 2010). The values given are normalized averages (normalized to 1 GWe of installed capacity) from the 22 units analysed:

1. *Employment*. Building a new nuclear power plant will result in the creation of 1400 to 1800 jobs, with peak employment as high as 2400 jobs. During operation each unit generates from 400 to 700 permanent jobs, receiving salaries 36% higher than existing average local salaries. Such an increase in population may generate an equivalent number of local jobs for the goods and services needed.
2. *Local economic benefits*. The operating plant will require direct goods and services for some $430 million in the local communities and $40 million for labour income, to which indirect effects, amounting to some 7%, have to be added.
3. *Federal, state and local taxes*. On average, federal tax payments will amount to $75 million, while provincial and local tax revenue will amount to $20 million per year. These taxes are used to create state and local infrastructure.
4. *New plant construction*. The construction of a new nuclear power plant boosts the supply of commodities such as concrete and steel, and hundreds of components and services, such as transportation. It has been estimated that a new nuclear power plant may need some 1 million m^3 of concrete and 66,000 tons of steel, 70 km of piping, 480 km of wiring, and 130,000 electrical components. Although the major nuclear components come from other places, many items could be provided within the area of influence.

Spain has also developed studies on the local socio-economic impacts of nuclear power plants. A study conducted by the Burgos University

Department of Economics at the Nuclenor-owned Garoña plant (a 460 MWe GE-BWR) concluded that between 1992 and 2006 the plant spent €80 million on local goods and services, and provided local tax revenues amounting to €6 million. The plant also invested about €16 million per year in plant retrofitting, technological innovation, and research and development (NUCLENOR, 2007). The study makes an analysis of the social and economic evolution of the population within a 35 km radius of the plant, from before construction to 2007, i.e. after 37 years of operation, by measuring unemployment, commercial activities, industrial development, financial entities and the increase of cars and telephones. To separate the impact from the plant, the results have been compared with the corresponding average numbers for the rest of the provincial territory. Factors of 2 have been found for some of the indexes.

Another study was conducted for the Ascó and Vandellós plants operated by ANAV (which include three W-PWR, of one GWe each). The study was undertaken by the Rovira i Virgili University (URV) in the city of Tarragona using the output–input methodology described above (ANAV, 2011). It found that output to the Catalonian economy from the activities of ANAV and its workers is four times the initial input during the five-year period from 2004 to 2008. This factor is 3.3 when the territory is limited to the province of Tarragona, where the plants are located. Similar results are obtained for the employment created.

8.6 Risks and detriments of nuclear energy

The risks and detriments involved in nuclear power development come from three major characteristics of nuclear energy: it is energy intensive, it generates radioactive material and it generates strategic material. To avoid damaging accidents, the chain reaction, the transfer of heat and the confinement of radioactivity have to be kept under strict control by nuclear safety measures. To protect the health and safety of workers, the general public and the environment, radioactive materials have to be confined for lengths of time well beyond when they were first produced, requiring the establishment of a radioactive waste management system and a final repository strategy. The production of strategic materials creates the risk of nuclear weapons proliferation, which needs to be controlled by tough security measures. As a large industrial installation, nuclear power plants also create non-radiological detriments: there is an increase in light and heavy traffic, noise, pollution, aesthetic impacts and heat releases. To a greater or lesser degree, some of these aspects may affect the whole world, the country where the nuclear power is developed, and/or the region where nuclear power plants are operated.

8.6.1 Nuclear safety and the perception of risk

A nuclear reactor core has to maintain two equilibria: the rate of neutrons produced has to be equal to the rate of neutrons lost, and the rate of heat generated has to be equal to the rate of heat removed. Both equilibria are linked through so-called reactivity coefficients, the values of which make the system stable or not. Perturbations in these equilibria may be introduced by internal or external causes and by human error. Consequently, the design of a reactor has to contemplate all the possible inputs and determine how the plant will react against them in such a way that no damage to the core is produced and no release of radioactive products should occur; under these circumstances, the reactor is declared safe.

Nevertheless, accidents may occur when not all the potential inputs (or a combination of them) have been considered, when their magnitude has not been measured properly, when multiple human errors have been committed, or multiple safety equipment is unable to work due to common-cause failures. Combinations of all these are possible, although remote. When such accidents occur, there is a possibility that radioactive products will be released to the exterior, with the likelihood that the health and safety of people will be affected and that environmental contamination by radioactive products will occur.

The accident at TMI-2 was due to a combination of equipment failure, poor maintenance and human error, the accident at Chernobyl-4 was caused by human mismanagement of the reactor's unstable condition (INSAG, 1992a), whilst the accident at Fukushima-1 had its origin in the multiple common-cause failures produced by an earthquake and a following tsunami, against which the plant was not designed. In TMI-2 the release of radioactivity was limited and the health and safety of people was not at risk; in Chernobyl the release was very large and the radiological consequences very serious; in Fukushima-1, the release from the three affected units was about one-tenth that of Chernobyl but the radiological consequences were limited, due to an efficient emergency management.

Relevant lessons have been learned from such events and these lessons have served to improve the safety of present and future nuclear power plants. These accidents demonstrate that absolute safety is not achievable, and that there will always be a residual risk, although that it should be as low as possible, and acceptable. Prevention of accidents and mitigation of their consequences is the main aim of nuclear safety.

The safety level of nuclear installations and activities that give rise to radiation risks can be improved and maintained by following the IAEA Fundamental Safety Principles (IAEA, 2006), which provide the basis for the safety requirements and safety guides and programmes which have been developed by the IAEA as part of its safety standards activities.

These principles apply to a nuclear power plant in all modes of operation and to its entire fuel cycle installations and activities, such as transportation of radioactive waste and nuclear materials, and their final disposal. The principles recommend the creation of a series of administrative envelopes and technical barriers that prevent accidents and mitigate their consequences.

These administrative barriers assign the prime responsibility for safety to the licensee, as well as allocating to government the responsibility of enacting a complete and satisfactory legal and licensing system and the creation of a regulatory body with three major activities: the development of a consistent set of safety standards, the verification of compliance with applicable standards, and an enforcement authority to correct any deviation. The licensee is also obliged to develop leadership and management for safety, based on the promotion of a safety culture within the installations and all related activities, on the regular assessment of safety performance and on feedback from operating experience. These administrative and procedural barriers are essential to achieve and maintain the required safety levels.

Technical barriers also help to prevent accidents and mitigate their consequences by adhering to the concept of defence-in-depth, through a combination of consecutive and independent levels of protection which would have to fail to cause the release of radioactivity to the environment. Such levels include conservative designs and use of materials of high quality and reliability; the introduction of control, limiting, protection and monitoring systems; the addition of technological safeguard systems to cope with accidental situations and to mitigate their consequences, including so-called passive systems; the application of well-developed and trained emergency procedures; and the availability of emergency measures to protect people outside.

The safety level of a nuclear power plant can also be measured through its complementary function: risk. A quantitative risk assessment methodology for nuclear power plants was first introduced in 1975 by the *Reactor Safety Study* (NRC, 1975); the methodology was later repeated in Germany and an English translation produced (EPRI, 1981) and later refined to consider five specific nuclear power plants covering the nuclear technologies used in the USA (NRC, 1990). This new methodology has been used widely (although covering only the first two levels of these studies) across practically all nuclear power plants in the world.

Such 'Probabilistic Safety Analyses' (PSAs) are divided into three levels. Level 1 PSA determines the expected frequency of accidents producing core damages, the values obtained ranging from 1 in 10,000 to 1 in 100,000 per year and reactor. Level 2 PSA estimates the conditional probability of an early release of radioactive products by failure of the containment system within a damage core, with values found to vary from 1 in 10 cases to 1 in 100 cases. The accepted recommended values (INSAG, 1992b) are less than 1 in 100,000 per year and reactor for Level 1, and a conditional probability of 1 in 10 cases for Level 2 PSA. Level 3 PSA determines the complementary distribution function of the radiological consequences to the health and safety of the public and also the economic consequences derived from losing the plant and restoring the environment. When these results are compared with other technological risks and those from natural events, it is concluded that the risks of nuclear power plants are several orders of magnitude lower.

Although it may seem sufficient to estimate nuclear risks and put absolute, and also relative, values on acceptable risks, risks perceived by the individual and society are also a reality to be considered. In an analysis of perceived risks, it is necessary to consider, among the major aspects, the benefits obtained, familiarity with the type of risk, and the nature and time dependence of the harm produced.

The benefits obtained determine the perception of risk. Individuals accept high risks when the benefits are clear to them, which explains the acceptance of driving a car or smoking. The benefits from nuclear power are not clearly estimated by individuals and society: the need and appreciation of the benefits obtained from electricity are well understood, but electricity can also be provided by other means. Because of this, it is necessary to explain, once again, the worldwide, national and local socio-economic benefits coming from nuclear power. To make the picture complete, it would be necessary to compare the risks and benefits of the other sources generating electricity, but such an analysis is outside the scope of this chapter.

Familiarity with the nature of a risk and its frequency is another major ingredient in the perception of risk. Although the use of radiation is now part of everyday life for many people, mainly through its use in medicine, fear of radiation is very high due to its peculiar nature, which is difficult to understand. An average individual receives doses, for medical purposes, which are much larger than those from natural radiation and orders of magnitude larger that those the most exposed person will receive from the operation of nuclear power plants and related fuel cycle activities. Nevertheless, society grossly exaggerates the risks perceived from nuclear power, despite the efforts made and the evidence presented to explain the real situation.

The timing of the harm produced is another aspect of interest. When damage done shows immediately, the perception of risk is different from when it may or may not come later in the life of the person, or if the risks may still exist for future generations. It is well known that high radiation doses may produce deterministic effects and that damage will show up soon after exposure but, most frequently, even in the case of severe accidents, most exposures produce low doses with the potential to produce stochastic effects, sometimes many years later. These circumstances have produced a considerable increase in perceived risk, with people believing that any exposure, however low, will produce the expected effects with certainty, despite efforts made to show that the probability of the effect is very low and proportional to the dose received.

Despite efforts made to increase safety, accidents cannot be completely discounted and preparation for them should be in place. Two instruments have been created to protect the health and safety of the public, and to protect private and public properties and the environment. The first is a legal instrument based on the concept of third-party liability for the damage caused. The second is the preparation and maintenance of emergency procedures and the corresponding equipment needed to protect the health and safety of the individuals affected.

8.6.2 Third-party liability: a legal solution to compensation

Third-party liability has been a major concern since the beginning of nuclear power development. US legislators recognized early that government-owned and privately owned nuclear facilities could be built and operated safely, but that zero risk could never be achieved. As early as 1954, work started to develop a bill on third-party liability, which was finally signed into law by President Eisenhower in 1957. The Price–Anderson law was a recognition of the rights of affected persons to be compensated for the harmful effects of radiation. The development of this interesting process is ably described by Mazuzan and Walker (1990). The law, still in force, has been amended several times, generally in the sense of increasing people's protection and introducing requirements to protect the environment. The law has been a model for other national and international developments.

The NEA/OECD was the first international institution to propose to its member states the creation of a Convention on Third Party Liability, which was first established in 1960 and later amended in 1964 and 1982 (NEA, 1982). The Paris Convention entered into force in 1988 and was ratified by 19 NEA member states. The Convention provides for compensation for injury or loss of life of any person, as well as for damage to or loss of any property caused by a nuclear accident in a nuclear installation, or during the transport of nuclear substances to and from installations. The Convention does not cover damage to the nuclear installation itself. The maximum liability of a nuclear installation operator amounts to 15 million Special Drawing Rights (SDR), an accounting unit created by the International Monetary Fund (IMF) to include key international currencies reviewed and adjusted every five years, the value of which is close to a US dollar. However, the NEA recommended that parties to the Paris Convention should set the maximum liability amount to not less than SDR150 million, and most have done so.

The 1986 Chernobyl-4 accident demonstrated the need to increase the amounts of liability and to enlarge the types of damage provided. In response to that need, in 2004 the contracting parties to the Paris Convention adopted the Protocol to Amend the Paris Convention (NEA, 2004), which is still pending ratification. In the new Protocol, the amount for nuclear power plants is increased to €700 million and licensees will still be required to financially secure their liability. Parties to the revised Paris Convention will also be required to ensure the payment of nuclear damage claims where a licensee's financial security is unavailable or insufficient, up to the liability specified in the Convention. The Protocol will add certain types of economic loss, the cost of measures to reinstate a significantly impaired environment, loss of income resulting from that impaired environment and the cost of preventive measures, including loss or damage caused by such measures.

The Vienna Convention on civil liability for nuclear damage entered into force in 1963 (IAEA, 1963) and it has also been amended on several occasions. At a diplomatic conference in 1997, delegates from over 80 Member States adopted a Protocol to amend the 1963 Vienna Convention and also adopted a Convention on Supplementary Compensation for Nuclear Damage. The Protocol sets the possible limit of the licensee's liability at not less than SDR300 million. The Convention on Supplementary Compensation defines additional amounts to be provided through contributions by party states on the basis of installed nuclear capacity and a UN rate of assessment. The Protocol contains a better definition of nuclear damage (now also addressing the concept of environmental damage and preventive measures), extended the geographical scope of the Vienna Convention, and enlarged the period during which claims may be brought for loss of life and personal injury.

The Paris and Vienna Conventions are the international instruments created to compensate for the risks derived from potential nuclear accidents in nuclear power plants and related fuel cycle facilities and practices. They are supported by sound technological and economic considerations on the harm that accidents may produce and how such harms must be compensated.

8.6.3 Protection and inconveniences provided by emergency planning

Emergency planning lies within the IAEA Fundamental Safety Principles. Technically it is also considered as the last level of protection against accidents with external radiological consequences. Principle 9 requires that 'Arrangements must be made for emergency preparedness and response for nuclear or radiation incidents'. Emergency planning is a responsibility of government; arrangements should secure adequate response at the local, regional, national and even international levels, when required.

The line of responsibility for taking urgent decisions needs to be defined well in advance. In nuclear power plants, emergencies generally start within the plant and, as demonstrated in Fukushima, nuclear emergencies can be started by natural or manmade emergencies. During an onsite phase, the licensee is responsible for actions taken under the supervision of the regulatory authority, following well-known and rehearsed procedures. An alert situation should be declared quickly when there is no evidence that the plant can be brought under control. At that moment, an offsite emergency plan is initiated and conducted under the responsibility and authority of local, regional and state authorities, as the case may be, with the advice of the regulatory authority and with help and information provided by the licensee.

Public protection actions may include shelter, evacuation, decontamination, medical treatment (when necessary) and prophylaxis activities, such as ingesting potassium iodide to block radioactive iodine from entering into the body. Arrangements should include well-trained human resources, emergency procedures and reliable equipment, suitable installations and services for evacuees. National and international information, as well as humanitarian help to the people affected, are required.

Any inconvenience that people may suffer from emergency protection activities has the advantage of avoiding or reducing radiation exposure. These inconveniences are related to the conduct of periodic drills which often involve the nearby population. In case of real emergencies, remaining under shelter and being evacuated or displaced for long periods of time (as in the cases of Chernobyl-4 and Fukushima-1) are problems that affected people have to suffer. Decontamination of affected buildings to allow the return of evacuees and the restoration of agricultural soils can take a long time and create inconveniences to people's lives. These remote circumstances should be compared with the benefits that the neighbouring population will receive with certainty.

8.6.4 Radioactive waste management as a concern for current and future generations

The generation of radionuclides is a major concern in the development of nuclear power. A nuclear power plant acts as a radioactivity amplifier, the generated radionuclides appearing in different wastes requiring specific management systems. The half-life of some of these nuclides is much longer than the life of the power plant, and therefore the management system selected has to provide protection for present and future generations.

Most of these radionuclides are generated inside the fuel matrix by neutron fission and activation. Neutron activation products are also generated within the coolant and the structural materials close to the reactor core. Fission is so highly energy-intensive that the fission of a gram of uranium-235 per day will generate a power of roughly 1 megawatt. Thus, the quantities of fission nuclides (the most abundant) are very small but, nevertheless, their radioactivity (in terms of number of disintegrations per second) is very high, and they are retained within the fuel matrix and fuel cladding.

Through the fission and activation processes, a large variety of radioisotopes from light to heavy elements accumulate in the fuel, the structural materials and the coolant, producing two types of waste: limited and controlled quantities of gases and liquids containing small amounts of certain nuclides which are released to the environment; and solidified wastes of low and medium specific radioactivity (disintegrations per unit of mass) containing radionuclides of short and medium half-life, which are stored in final surface repositories. Under normal operating conditions, the radioactivity included in these waste products is no greater than 2% of the total radioactivity inventory of the reactor.

The bulk of the radioactivity inventory is found in the used fuel. Used fuel elements constitute by far the greatest problem and can be considered as waste, in a so-called once-through fuel cycle, or as a large energy resource when used in a uranium-239/plutonium-239 closed cycle.

The radionuclides generated within metallic or concrete structures remain there and are only of concern at the time of dismantling the plant. Radionuclides in the coolant include mainly short half-life gases and medium half-life metallic isotopes from the corrosion of metallic surfaces containing the coolant. A small amount of noble gases, tritium and volatile elements (such as iodine, tellurium and cesium isotopes) may be released to the coolant through small defects in the fuel cladding. A very small fraction of the coolant and its radioactive contents escape containment, and the released nuclides are captured by high-efficiency filters in the containment ventilation system.

The radionuclides in the coolant are treated. In PWRs gases are easily separated and stored for decay and finally vented to the atmosphere before refuelling outages; only krypton-85 (a beta/gamma emitter with a half-life of 10.6 years and of limited radiological importance) remains after one year's decay in the gas storage tank. In BWRs, gases carried out of the core with the steam are finally channelled to a delay system of various days, where again krypton-85 is the major radionuclide, although some xenon-135 (a beta/gamma emitter with a half-life of 5.26 days) is also released. Small amounts of iodine isotopes, mainly iodine-131 (a beta/gamma emitter with a half-life of 8.06 days), may also be released to the atmosphere.

Metallic radionuclides (present in the coolant as ions, colloids or particles) are separated out by filtration, ion exchange or evaporation; filters, spent ion exchange resins and evaporation concentrates are added to a solidifying matrix, usually cement, and put into standardized 220-litre drums, the radionuclides in the waste remaining fixed in the solid matrix. These drums together with other solid wastes such as air filters, contaminated tools and radiation protection equipment constitute the major part of the radioactive waste generated from operation. On average, a 1 GWe PWR generates from two to three drums per day, while a BWR, in which the coolant is purer, generates from four to five drums per day.

Tritium is a fission product and is also produced by neutron activation of boron and lithium, coolant additives in PWRs. It is a weak pure beta emitter with a half-life of 12.26 years which is also found in nature. Tritium substitutes for hydrogen in the water molecule and cannot be separated by ion exchange or evaporation, and is eventually found in releases of surplus water into nearby water bodies. Such releases may also contain small quantities of other radionuclides, mainly coming from activation of corroded materials.

Gaseous releases and releases to water bodies are limited by the regulatory authority and made under strict controls. There have been reports of tritium being found in underground water coming from leakages in pipes; timely remedies have been put into practice.

In many countries, the doses received by most exposed individuals are limited to one-tenth of a millisievert/year, a very small fraction of the doses received from natural radiation. In practice, such doses are orders of magnitude smaller than the regulatory limit. Moreover, to further protect people, the territory around a plant is the subject of monitoring programmes to determine radioactivity in air, soil, surface and underground water and food products, for which thousands of samples are taken and analysed per year to determine their radioactive contents. European Union countries have to report the findings, which are made public, as a requirement of the Euratom Treaty.

On top of this, epidemiological studies are conducted to determine the health of the affected population and to find potential diseases which could be caused by emitted radionuclides. If epidemiological studies have not shown any evidence of such effects, it is therefore concluded that there are no reasons to be concerned about the correct management of environmental releases. If they were to appear, the monitoring systems would detect them before any harm is done.

Solid waste coming from operation has low and medium specific radioactivity and does not include the very long-life radionuclides; cesium-137 (a beta/gamma emitter with a half-life of 30 years) and strontium-90 (a beta emitter with a half-life of 28 years) are typical contaminants. Generally, the drums are temporarily stored on the plant premises to be removed to an above-surface final repository. Such repositories are designed to last for some 300 years after closing, after which time it is assumed that risks of radiation are acceptable and well below limits.

When used fuel is considered a waste, there should be also a management system for it. This type of waste is highly radioactive and includes many long-life radionuclides, mainly isotopes from transuranic elements, some of which, apart from being beta/gamma emitters, are also alpha emitters. A large fraction of the energy emitted by disintegration is converted into heat, which must be removed for a long time. This fact complicates the final disposal of such waste. The best solution seems to be deep geological repositories for which research efforts are been conducted in many countries, though only Finland and Sweden have advanced projects, while the development of the US Yucca Mountain commercial repository has been stopped. In the interim, spent fuel is stored in decay pools or in air-cooled dry containers within the plant premises. A central storage facility is in operation in the Netherlands and a similar project is well advanced in Spain.

Principle 7 of the IAEA Fundamental Safety Principles requires that 'People and the environment, present and future, must be protected against radiation risks'. As the risks from long-term radioactive waste repositories could span generations, 'subsequent generations have to be adequately protected without any need for them to take significant protective actions'. This creates the problem of protecting future generations from the risks of activities from which the benefits went to previous generations, which seems unethical. Nevertheless, future generations will also receive the benefits of the scientific and technical knowledge and development from previous generations as a basis for their own progress. Moreover, future generations may find better ways to manage nuclear waste.

In conclusion, the management of radioactive waste does not substantially increase the risk and detriments of the plant itself. The final repositories have to pay local taxes and provide subsidies which benefit the economic and social development of the affected population. The storage of used fuel in the plant premises increases the inventory of radioactivity in the plant, but the probability of releasing radioactivity from this source is much lower than from the reactor core. The transport of used fuel to a temporary or final repository, and the transport of low and intermediate waste from the plant to the repository, have impacts and inconveniences on traffic around the plant and along transport routes. The transport of such materials should comply with the transport requirements established by countries in accordance with the widely accepted IAEA transport requirements (IAEA, 1996), which help to ensure a very small risk of accidents.

8.6.5 Fear of radiation as a societal impediment

Ionizing radiation has shaped life on earth since it appeared and exposure to ionizing radiation will continue unabated. It comes from the earth and from the sky and there is no reason to avoid it, although protection from extended exposure to ultraviolet rays coming from the sun is recommended. Ionizing radiation is used widely and increasingly in medicine for both diagnosis and treatment; high levels of exposure are frequently delivered and there is no reason to avoid such uses, although valid efforts are made to reduce the doses and the risks of radiation accidents without reducing the benefits. Ionizing radiation also comes from nuclear power plants, in small amounts during normal operation and potentially in larger quantities in case of accidents; there is no reason to ban these plants, while increasing safety is a primary objective. Nevertheless, natural and medical radiation are socially acceptable, while radiation from nuclear power plants is grossly rejected. This rejection is supported by the precautionary principle 'when in doubt, keep it out'.

Protection against ionizing radiation is based on the principles of justification of practices, optimization of protection and limitation of individual dose and risks, which have been developed over time by the ICRP. These principles were introduced in international and national regulations, and are developed in Chapter 11 of this book.

Radiation protection principles are now included in the IAEA Safety Fundamentals (IAEA, 2006) and their meaning and application go beyond the intended ICRP thinking. The justification principle was initially intended for radiation use, though this chapter analyses its application to justify nuclear power. The optimization principle aims to provide 'the highest level of safety that can reasonably be achieved throughout the lifetime of the facility or activity, without unduly limiting its utilization'. When applied to a nuclear power plant and related fuel cycle installations and activities, this requirement should be applied to all modes of operation, from normal to accident conditions, and during the whole life of the installation.

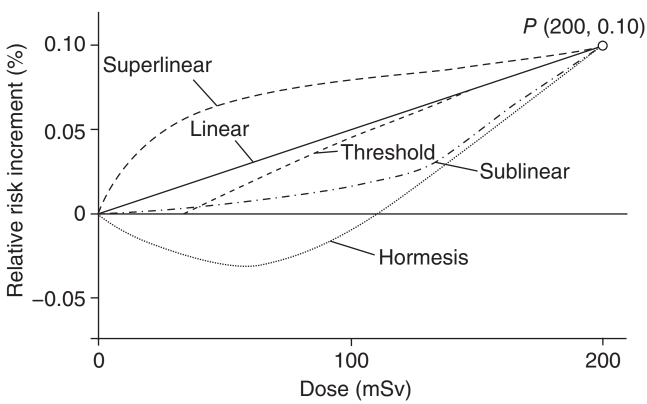
The limitation principle establishes that 'Measures for controlling radiation risks must ensure that no individual bears an unacceptable risk of harm'. In practice, this requirement translates to defining dose limits that should not be surpassed, and which are the upper legal bounds of acceptability, but which do not assure the best protection possible. When the limitation and optimization principles apply to any given installation or activity, the desired level of protection should be achieved without impairing the benefits that can be obtained from the installation or the activity in question.

Legal radiation dose limits have been suggested by the ICRP and accepted globally for a variety of circumstances and human organs. The limits for public exposures have been reduced to one millisievert/year, and in many countries lowered to one-tenth of a millisievert/year after applying the optimization principle. In practice, the dose received by the most exposed individual in the public population from a nuclear power plant is even less than that. These values should be compared with the average of 2.4 millisieverts/year that is received from natural radiation.

Although these values are well supported by scientific evidence, an intense 'radio phobia' has grown in developed countries. One of the reasons for such a situation is the so-called linear non-threshold approximation (LNT) which assumes that any radiation dose may produce harm. Other proposed approximations are included in Fig. 8.2.

The ICRP included the LNT hypothesis very early in its recommendations. CRP Publication 9 (ICRP, 1966) stated: 'Because of the lack of knowledge of the nature of the dose–effect relationship in man, the Commission sees no practical alternative, for the purpose of radiological protection, to assuming a linear relationship between dose and effect'. The Commission understands that the assumption of no threshold may be incorrect, but it is satisfied that application of LNT will probably not underestimate radiation risks.

Although much has been learned about carcinogenesis, the fact is that the LNT hypothesis has continued to be used since that time. The situation has been reviewed in subsequent ICRP reports, mainly in ICRP Publication 26 (ICRP, 1977) and ICRP Publication 60 (ICRP, 1990), although without change. In fact, the dose limits have been reduced because the risk coeffi-



8.2 Relative risk increment of stochastic effects for low radiation doses for different hypotheses: (1) linear non-threshold recommendation; (2) supralinear assumption; (3) sublinear assumption; (4) threshold hypothesis; (5) hormesis (beneficial) effect.

cients in the LNT hypothesis have been increased. In ICRP Publication 103 (ICRP, 2007) a certain concern has been expressed when recognizing that the LNT hypothesis makes it 'impossible to derive a clear distinction between safe and dangerous'. Moreover, it advises that the LNT hypothesis should only be used for the optimization process and not for epidemiological studies, as it frequently is.

The ICRP recommendations have been accepted by international organizations, mainly the IAEA, World Health Organization (WHO) and others, and have been introduced in national, supranational (EU) and international regulations and standards, not always concurrent with the understandings defined by the Commission. As such, they are the main source of social and political concerns. Over time, what started as a working assumption has come to be considered as a scientifically documented fact by the public and mass media, and even by some regulatory bodies and many pro-LNT scientists.

'Radio phobia' against nuclear power started in the early 1970s, gaining impetus in the late 1970s, and started to produce serious consequences during the 1980s after the TMI-2 (1979) and Chernobyl-4 (1986) accidents, the consequences of which were grossly amplified by the media, some political parties and non-governmental organizations. The Fukushima large earthquake and tsunami-driven events have again increased public radio phobia, and increased serious concerns within international institutions and national authorities about the safety of currently operating nuclear power plants, which are now being reviewed.

The application of the LNT model to estimate the stochastic consequences of small doses and small dose rates received by a large number of people as a consequence of the Chernobyl accident have not so far been proven by direct observation. The existence of a hormetic, i.e. beneficial, relationship between low doses and consequences has not yet been scientifically proven for all cases and circumstances, despite the intense research on the matter which has been conducted (DOE, 1998).

It has to be recognized that the LNT hypothesis may have provided a reasonably conservative approach to radiation risk assessment, but its introduction into the national and international regulatory system has probably gone too far. It has contributed to create an intense social and political radio phobia which has clearly limited the beneficial application of nuclear energy.

In conclusion, neighbourhood populations may be exposed to tiny radiation doses, smaller than the ones received from natural radiation, from gaseous and liquid releases, which are limited and well controlled in the origin of the release and in the surroundings through an extensive monitoring programme. Moreover, epidemiological studies have not revealed any harm from such radiation. The concerns raised by these doses are not founded.

8.6.6 Security as a subject of increasing relevance

Nuclear installations and relevant activities may be the objects of terrorist attacks bacause of the potentially large social distortion they could produce. The design and operation of nuclear installations include very strict access control, sophisticated intruder detectors, entrance delay technologies and armed police forces. Operation now also involves intelligence and external help. Nuclear installations are generally very robust for safety reasons and there is a synergy between safety and security, which has recently been analysed by the INSAG (INSAG, 2010) and by the IAEA AdSec Group (IAEA, 2007). A justification document should include references to national and international regulation use in the design and organization of the security system, though it will be necesary to reserve details (for security reasons).

8.6.7 The risks of nuclear weapons proliferation: a major world concern

Strategic materials for nuclear weapons proliferation can be obtained from the nuclear fuel cycle. These materials can be uranium-235 from the first part of the cycle, or manmade plutonium-239 produced in the reactor and separated in the reprocessing side of the fuel cycle. Once the strategic materials are available, the design, construction and deployment of nuclear weapons is, although very costly, not a big technical problem, and is one which could be mastered by many. Light and heavy water reactors are the ones in operation now and are the candidates for short- and medium-term deployment, so the following considerations are limited to such reactor models; the future use of fast breeder reactors using the uranium-238/plutonium-239 fuel cycle will need further considerations, which are outside the remit of this chapter.

Light water reactors use natural uranium enriched in uranium-235 from its natural value of 0.7% up to 3–5%. Enrichment is a well-known physical isotope separation process achieved by gas diffusion through membranes or by ultracentrifugation; laser separation is now laboratory proven but not yet deployed on a commercial scale. The most available and economical way to produce enriched uranium is by ultracentrifugation, and the same equipment can serve to reach reactor enrichment levels of up to 90% weapons-grade enrichment. This process can be fully achieved by any reasonably developed state wanting to follow this path.

Plutonium-239 is an activation product of uranium-238 present in the reactor, which accumulates slowly within the fuel matrix; over time, other isotopes of plutonium which are not fissile start to accumulate (mainly plutonium-240). Weapons-grade plutonium-239 is found when irradiation times are very short, of the order of a few months. The fuel cycle in a power plant is much longer than that, and therefore the plutonium produced (called reactor-grade plutonium) is contaminated with the other isotopes, although it could also serve to produce lower-yield nuclear weapons. In any case, the plutonium produced has to be separated from the other components, uranium-238 and unburnt uranium-235 and fission products. This separation can be easily achieved by a well-known chemical process called PUREX which is easily accessible, although the presence of the highly radioactive fission process complicates the chemical separation. A 1 GWe LWR needs some 20 tonnes of fuel per year and generates about 200 kg of reactor-grade plutonium.

Efforts have already been made (and new solutions are being developed) to make the fuel cycle proliferation-proof against any desire to use strategic materials for non-peaceful purposes. Although limited advances have been achieved, it is clear that there will not be any easy technical solutions to avoid proliferation; only policy and diplomacy can serve to reduce the risks, as explained by the Nobel Prize winner, Burton Richter (Richter, 2008). The Non Proliferation Treaty (NPT) and corresponding safeguards under the control of the IAEA are the diplomatic instruments that have been created to reduce such risks. This matter is considered in depth in Chapter 13 of this book.

Although very effective, the NPT and its safeguard requirements cannot be absolutely proliferation proof. Policy proposals have been formulated within the IAEA and leading countries to internationalize the fuel cycle by providing enrichment and reprocessing services by the most developed countries under special international controls. The Global Nuclear Energy Partnership (GNEP) proposed by former President Bush is another example. Both initiatives have not yet been fully developed. All these technical, political and diplomatic efforts have considerable potential to reduce, but not completely eliminate, the risk of proliferation.

The risk of proliferation is highly dependent on the geopolitical confrontations within the world and is therefore difficult to evaluate. But the culprit is not the peaceful uses of nuclear power: in fact, the production of strategic materials is cheaper and more effective if undertaken at dedicated installations. In practice, effective technologies, inspections and treaties will reduce the risks of proliferation as far as possible, and this matter should not be a deterrent for the peaceful development of nuclear energy.

8.6.8 Condenser heat rejection: the major non-radiological detriment

As with any other large industrial installation, a nuclear power plant, fuel cycle facilities and associated activities produce substantial non-radiological physical and chemical impacts on their surroundings during pre-construction, construction, operation and dismantling. These impacts are generally considered in an environmental report, a licensing requirement in most countries. The USA NRC requirements are among the most developed regulations, and they also include economic and radiological impacts (NRC, 1984).

Nuclear regulatory authorities are not the only regulators intervening in this process; other local, regional and state authorities also participate in the review, generally in a coordinated manner, to verify compliance with other regulations on environmental protection, such as those related to air and water quality and on land use.

For non-radiological impacts, any environmental impact study starts by describing the affected territory and its current use, industrial and recreational development; the surface and ground water hydrology and the use given to such waters; the meteorology and air quality, and the terrestrial and aquatic ecology, amongst the major aspects. With this knowledge, the impacts during pre-construction and construction activities are analysed, generally divided into three levels of significance: small, moderate, or large. For all these impacts, mitigation measures are also considered, though, in such an analysis, unavoidable adverse environmental impacts can be found for which no practical means of mitigation are available.

During pre-construction and construction activities, the major unavoidable environmental impact would be the land to be occupied by the plant buildings and the land used temporarily for construction purposes; additional land will also be needed to build new or widen existing roads and electrical energy corridors. The high energy intensity of nuclear power does not need additional land for storing new and used fuel, as is the case for fossil fuels (mainly coal). Considerations should also be given, among other things, to the use of water and construction materials; the effects that excavation and dewatering will produce on groundwater aquifers; the ecological impact on terrestrial and aquatic losses; and the increase of traffic and health effects due to fugitive dust, noise and transportation. The purpose of these considerations is to evaluate them and to define mitigation and controls aimed at lessening the adverse impacts.

The land used for buildings will be considerably improved by trees, gardens and lawns. When the plant ends its useful lifetime, decommissioning will restore the site and make it useful for other purposes (decommissioning is considered in detail in Chapter 24 of this book). The operation of nuclear power plants is very clean; negligible amounts of conventional pollutants are released and solid conventional waste is very limited. Non-radiological health impacts to members of the public, including etiological agents, noise, electromagnetic fields, occupational health, and transportation of materials and personnel are minimal and well controlled to verify compliance with applicable regulations.

During operation, the major non-radiological impacts are the use of water to cool the condenser and the effects that heat rejection produces in the affected water bodies. Nuclear power plants have a low thermal efficiency of about one-third; therefore two-thirds of the heat produced in the reactor core by fission is waste heat that has to be rejected to water bodies and eventually to the atmosphere. Some 5% of this heat is released within the plant itself; therefore some 62% of the generated heat has to be removed by the condenser water. Large quantities of water have to pass through the condenser to remove such heat, and the flow of water depends on the limit put into the outlet temperature; for a 10°C increase, in 1 GWe plant, some 45 m^3/s is needed.

To achieve heat rejection, two types of systems have been developed. Once-through systems take water from a large water body and discharge it to the same water body at a higher temperature and at a different point. Only plants built in coastal locations and in the proximity of large rivers or lakes can use such systems. In closed circuit systems, the warmed water is cooled in a cooling tower or spray pond and recirculated through the condenser. In this way, the waste heat is finally deposited into the air. Combinations of once-through and closed circuits are frequently found, the once-through systems being used when temperature limitations in the receiving water body can be complied with (mainly during the winter), and closed circuits otherwise.

Once-through systems may cause damage to living organisms in the water body as a result of changes in temperature, the impingement of larger organisms in the water-intake screens and the entrainment of smaller organisms that pass through the condenser. Deleterious effects may also be produced by the use of chlorine and biocides, by changes in the water quality, mainly oxygen content and increases in salinity. All these effects have the potential of introducing changes into the aquatic ecosystem; they should be known to verify compliance with environmental regulations.

A large variety of chemicals are added to wet cooling towers to control bacteria and prevent corrosion. Such chemicals might eventually be discharged to an adjacent water body when recirculation water retaining impurities is taken out of the tower; this is called a blowdown. Blowdowns have to be controlled and should not be discharged to public waters without treatment and control in accordance with regulations. Such chemicals may also escape to the atmosphere with small water droplets in the drift, i.e. steam released from the cooling towers; these chemicals will be deposited and will accumulate on surfaces near the plant. The effects of this have to be controlled; in modern cooling towers, drift is limited and the effects minimal.

Although non-radiological effects are well recognized, their study and quantification constitute the basis of their mitigation, until compliance with existing regulations. In general, nuclear power plants create a clean environment with limited physical and ecological impacts.

8.7 Conclusions

The justification principle has not been systematically developed for application to nuclear power plants, fuel cycle facilities and related relevant activities, despite the fact that it is a key requirement in international and supranational regulatory activities. Only the UK has so far developed regulations and guidance for justification of nuclear energy, now being applied in the justification of some new nuclear designs. Other countries have developed detailed regulations regarding environmental analysis which also include social and economic aspects and are close to justification exercises. Most frequently, economic advantages and benefits are the only basis for decisions.

The application of the justification principle, as defined in the IAEA Fundamental Safety Principles, and within a well-defined and complete process, will serve to present to society a valid account of the benefits derived from nuclear energy and the risks and detriments associated with it. These studies will facilitate public understanding and help in decision processes.

There are many examples regarding nuclear installations and relevant related activities where justification could provide valid insights to high-level decision-making processes. The elements to be taken into account, the justification process itself, the definition of a justification authority, and the value and limitations of the justification decision, all need to be defined formally. Valid tools are already available to define and quantify some of the key elements which are part of the justification equation; others need further research and development.

8.8 References

ANAV (2011), *Impacto económico de ANAV en el territorio*, Departamento de Economía, URV, Tarragona, España.

CCC (2011), *The renewable energy review*, Committee on Climate Change, London, CCC.

DOE (1998), *Biological Effects of Low Dose and Dose Rate Radiation. Research Programme*, Washington DC, US Department of Energy.

EPRI (1981), *German risk study – Main report. A study of the risk due to accidents in nuclear power plants*, Electric Power Research Institute, EPRI NP-1804-SR Special Report, Palo Alto, CA.

Exelon (2008), *The impact of Exelon's proposed construction and operation of a nuclear power facility on business activities in Victoria County and Texas*, The Perryman Group, Valley Mills, Waco, TX.

FISA (2001), *EU Research in reactor safety. Proceedings*, Directorate General for Research, EUR 20281, Brussels, European Communities.

FISA (2003), *EU Research in reactor safety. Proceedings*, Directorate General for Research, EUR 21026, Luxembourg, European Communities.

FISA (2006), *EU Research and training in reactor systems. Proceedings*, Directorate General for Research, EUR 21231, Brussels, European Communities.
IAEA (1963), *Vienna Convention on civil liability for nuclear damage*, Vienna, IAEA.
IAEA (1996), *Regulations for the safe transport of radioactive materials*, IAEA Safety Standards Series, Requirements, No. ST-1, Vienna, IAEA.
IAEA (2006), *Fundamental Safety Principles*, SF-1, Vienna, IAEA.
IAEA (2007), *Engineering safety aspects of the protection of nuclear power plants against sabotage*, IAEA Nuclear Security Series No. 3, Vienna, IAEA.
ICRP (1966), *Recommendations of the International Commission on Radiological Protection*, ICRP Publication 9, Pergamon Press, Oxford, UK.
ICRP (1977), *Recommendations of the International Commission on Radiological Protection*, ICRP Publication 26, Ann. ICRP 1(3).
ICRP (1990), *1990 Recommendations of the International Commission on Radiological Protection*, ICRP Publication 60, Oxford, New York, Frankfurt, Seoul, Sydney, Tokyo, Pergamon Press.
ICRP (2007), *The 2007 Recommendations of the International Commission on Radiological Protection*, ICRP Publication 103, Elsevier.
INSAG (1992a), *The Chernobyl accident:Updating of INSAG-1*, INSAG-7, IAEA Safety Series, Vienna, IAEA.
INSAG (1992b), *Probabilistic safety assessment*, INSAG-6, IAEA Safety Series, Vienna, IAEA.
INSAG (2003), *Maintaining knowledge, training and infrastructure for research development in nuclear safety*, INSAG-16, Vienna, IAEA.
INSAG (2006), *Stakeholder involvement in nuclear safety issues*, INSAG-20, Vienna, IAEA.
INSAG (2008), *Nuclear safety infrastructure for a national nuclear power programme supported by the IAEA Fundamental Safety Principles*, INSAG-22, Vienna, IAEA.
INSAG (2010), *The interface between safety and security at nuclear power plants*, INSAG-24, Vienna, IAEA.
Mazuzan G T and Walker J S (1990), *Controlling the Atom. The Beginnings of Nuclear Regulation 1946–1962*, University of California Press, Berkeley, Los Angeles, London.
NEA (1982), *Convention on Third Party Liability in the field of nuclear energy of 29th January 1960, as amended by the Additional Protocol of 28th January 1964 and by the Protocol of 16th November 1982*, Paris, Atomic Energy Agency.
NEA (2004), *Protocol to amend the Convention on Third Party Liability in the field of nuclear energy of 29th January 1960, as amended by the Additional Protocol of 28th January 1964 and by the Protocol of 16th November 1982*, Paris, Atomic Energy Agency.
NEA (2008), *Nuclear Energy Outlook, 2008*, NEA No. 6348, OECD Nuclear Energy Agency, Paris.
NEI (2008), *Economic benefits of the North Anna power station. An economic impact study by the Nuclear Energy Institute and Dominion*, Washington DC, NEI.
NEI (2010), *Nuclear power plants contribute significantly to state and local economies*, Washington DC, NEI.
NIA (2008), *Application to Justify New Nuclear Power Stations*, UK Nuclear Industry Association, London, NIA.

NRC (1975), *Reactor Safety Study. An assessment of accident risks in U.S. commercial nuclear power plants*, United States Nuclear Regulatory Commission, WASH-1400 (NUREG-75/014), Washington DC, NRC.

NRC (1984), *Environmental protection regulations for domestic licensing and related regulatory functions*, Nuclear Regulatory Commission 10 CFR Part 51, Washington DC, NRC.

NRC (1990), *Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants*, NUREG-1150 (1990), Washington DC, NRC.

NUCLENOR (2007), *Estudio sobre la incidencia económica y social de la central nuclear de Garoña. Informe 2007*, 2a edición, Fundación General de la Universidad de Burgos, Burgos, España.

OECD (2011), *Projected costs of generating electricity in 2005–2010*, Nuclear Energy Agency/International Energy Agency, Paris.

PG&E (2010), *The local economic impacts of decommissioning the Diablo Canyon Power Plant*, Orfalea College of Business, California Polytechnic State University, San Luis Obispo, CA.

Richter B (2008), *Reducing proliferation risks, Issues in science and technology. Fall 2008*, National Academy of Sciences, Washington DC.

Richter B (2010), *Beyond Smoke and Mirrors: Climate Change and Energy in the 21st Century*, Cambridge University Press, Cambridge, UK.

Spadaro J, Langlois L and Hamilton B (2000), *Assessing the Difference: Greenhouse Gas Emissions of Electricity Generation Chains*, IAEA Bulletin 42:2, IAEA, Vienna (http://www.iaea.org/Publications/Magazines/Bulletin/Bull422/article4.pdf).

UIC (2001). *Greenhouse Gas Emissions from the Nuclear Fuel Cycle* (http://world-nuclear.org/co2&nfc.htm).

UK (2004), *The Justification of Practices Involving Ionising Radiation Regulations 2004*, No. 1769, Stationery Office, London, Health and Safety, Environmental Protection.

UK (2008), *Guidance for Applications Relating to New Nuclear Power*, Stationery Office, London, Department for Business, Enterprise and Regulatory Reform, BERR.

UN (1998), *Kyoto Protocol to the United Nations Framework Convention on Climate Change*, United Nations, New York.

University of Nevada (2003), *The economic impact of the Yucca Mountain nuclear waste repository on the economy of Nevada*, Centre for Business and Economics Research, University of Nevada, Las Vegas, NV.

Weart R S (2008), *The Discovery of Global Warming*, revised and expanded edition, Harvard University Press, Cambridge, MA, and London.

WNA (2011), *The Economics of Nuclear Power (updated 9 March, 2011)*, World Nuclear Association, London.

9
Available and advanced nuclear technologies for nuclear power programs

S. BILBAO Y LEÓN, Virginia Commonwealth University, USA
and J. H. CHOI, J. CLEVELAND, I. KHAMIS, A. RAO,
A. STANCULESCU, H. SUBKI and B. TYOBEKA,
International Atomic Energy Agency (IAEA), Austria

Abstract: This chapter discusses the various nuclear technologies currently available for near-term deployment, as well as those in advanced stages of development that are expected to become available in the near to medium term. The chapter includes a brief overview of innovative nuclear technologies proposed for the longer term. Finally, the chapter offers some insights about the use of advanced nuclear technologies for non-electrical applications.

Key words: advanced nuclear reactor designs, evolutionary nuclear reactor designs, innovative nuclear reactor designs.

9.1 Introduction

In addition to the support required in the development of the infrastructure necessary to deploy a new nuclear program, newcomer countries have also indicated a desire to receive guidance in the process of evaluating the different nuclear technology options.

Countries, both those considering their first nuclear power plant and those with an existing nuclear power program, are interested in having ready access to the most up-to-date information about all available nuclear reactor designs as well as important development trends. To meet this need, the International Atomic Energy Agency (IAEA) has developed the Advanced Reactors Information System (ARIS) (IAEA, 2010), a web-accessible database that provides Member States with balanced, comprehensive and always up-to-date information about all advanced reactor designs and concepts.

In addition to having accurate information about the various nuclear technologies available, the key technical characteristics of a particular nuclear project should be clearly understood and specified at the onset of

Published by Woodhead Publishing Limited, 2012

the project. In this way both the technical and economic benefits of the alternative nuclear power plant designs and associated technologies can be objectively assessed against the situation and the needs of each country, and the most suitable design can be selected. Nations need to follow a design-neutral systematic approach that evaluates the technical merits of the various nuclear power plant technologies available on the market based on each country's needs and requirements.

The objective of this chapter is to help the reader differentiate among the different kinds of nuclear reactors and develop a clear picture about the current status of nuclear power technology. The chapter describes in some detail the most relevant nuclear reactor designs developed by all the suppliers/designer organizations in the world, highlights their advantages and disadvantages, and provides an update about the status of development and deployment of each one of them.

Because nuclear technology can also be used for many applications in addition to the production of electricity, and because many newcomer countries are interested in these non-electric applications almost as much as they are in the production of nuclear electricity, the chapter also provides a summary of the various non-electric applications of nuclear power and the technology needed to effectively deploy them.

9.2 Classification of advanced nuclear reactors

IAEA (1997a) defines advanced nuclear plant designs as those designs of current interest for which improvements over their predecessors and/or existing designs are expected. Depending on the amount of modifications implemented, advanced reactor designs can be categorized evolutionary or innovative. An evolutionary design is an advanced design that achieves improvements over existing designs through small to moderate modifications, with a strong emphasis on maintaining design proveness to minimize technological risks. The development of an evolutionary design requires at most engineering and confirmatory testing. An innovative design is an advanced design which incorporates radical conceptual changes in design approaches or system configuration in comparison with existing practice. Substantial research and development efforts, feasibility tests, and a prototype or demonstration plant are required prior to the commercial deployment of this type of design.

An alternative classification was coined by the Generation IV International Forum GIF (2002), which divided nuclear reactor designs in four generations. The first generation consisted of the early prototype reactors of the 1950s and 1960s. The second generation is largely made up by the commercial power plants built since the 1970s and that are still operating today. The

Generation III reactors have been developed in the 1990s and include a number of evolutionary designs that offer improved performance, safety and economics. After the increased interest in nuclear power seen in the first decade of the twenty-first century, additional improvements are being incorporated into Generation III designs, resulting in several concepts that are actively under development and seriously considered for near-term deployment in various countries. The Generation III designs loosely correspond to what in the ARIS system are called evolutionary designs (IAEA, 2010) and it is expected that they will constitute the bulk of the new nuclear plants built between now and 2030. Beyond 2030, it is anticipated that new reactor designs will address key issues such as the closure of the fuel cycle or proliferation concerns while possibly ensuring competitive economics, safety and performance. This generation of designs, the Generation IV, consists of innovative concepts in which substantial development is still needed.

Traditionally, nuclear reactors have been classified depending on the energy of the neutron spectrum they use to produce the fission in the fuel, or depending of the coolant they use to extract the fission energy from the core. With regard to the first criterion, nuclear reactors can be thermal when using low energy neutrons and fast when using much higher energy neutrons that are not slowed down by a moderator. With regard to the coolant, nuclear reactors can be classified as water-cooled reactors (WCR), gas-cooled reactors (GCR), liquid metal-cooled reactors (LMR) and molten salt-cooled reactors (MSR). Water-cooled reactors at the same time can be classified as boiling water reactors (BWR), in which the core is at relatively low pressure and the coolant is allowed to boil; and pressurized water reactors (PWR), in which the core is kept at high pressure and the coolant remains in a liquid state. Water-cooled reactors can also be divided into light water reactors (LWR) and heavy water reactors (HWR) that use deuterium water. While most HWRs belong to the pressurized water reactor type, and are also termed pressurized heavy water reactors (PHWR), some advanced designs use the boiling water reactor concept. As will be seen in upcoming sections, several advanced designs are what is called (IAEA, 1997a) an integral design, which refers to a reactor design in which the whole reactor primary circuit, including, for instance, pressurizer, coolant pumps, and steam generators/heat exchangers, as applicable, is enclosed in the reactor vessel. Finally, depending on the size of the plant, nuclear designs can be classified (IAEA, 1999a) as small (less than 300 MWe), medium (between 300 and 700 MWe) and large (more than 700 MWe). Although innovative reactor designs do not always fit the following norm, in general it can be said that most water-cooled reactors and gas-cooled reactors are thermal reactors, while most fast reactors are cooled by liquid metals or molten salts.

9.3 Key advances in technology

Various organizations, including design organizations, utilities, universities, national laboratories, and research institutes, are involved in the development of advanced nuclear plants. The IAEA ARIS database (IAEA, 2010) summarizes global trends in advanced reactor designs and technology and provides balanced and objective information about all available designs.

Since this chapter focuses on the technology options that are available for newcomer countries, it will concentrate on the evolutionary reactor designs, as these are the most likely candidate technologies for most countries' first nuclear power plant, particularly in the near to middle term. For completeness, however, there will also be included a brief discussion examining future trends for the development of nuclear reactors in the long term.

Evolutionary reactor designs have concentrated on improving the economics and the performance of existing nuclear reactors. At the same time, these designs meet even more demanding nuclear safety requirements than those currently in operation. While efforts have also been made in optimizing the use of fissionable materials and minimizing the production of used fuel and nuclear waste, it is expected that the closure of the nuclear fuel cycle will only be achieved once innovative reactor designs come online.

9.3.1 Trends in evolutionary plant design

One of the key objectives in the design of evolutionary reactors has been to reduce the *total* '*overnight*' *capital cost*[1] of a new nuclear power plant. To this extent, most designs include a significant reduction in the total number of structures, systems and components as well as a *simplification* of plant systems and components by using fewer and larger components, and by combining or eliminating functions or systems. The development of *standardized designs* that need to be validated and licensed only once also offers significant cost savings by spreading fixed costs over several units of the same standard design (IAEA, 1999b). Because first-of-a-kind reactor designs or plant components require detailed safety cases and licensing processes that result in major expenditures before any revenue is realized, standardization of a design is therefore a vitally important component of capital cost reduction. Many evolutionary designs have been developed based on 'user requirements', most notably by the Electric Power Research Institute (EPRI, 1995, 1999) and in the European Utilities Requirements (EUR, 2001), that is, the lessons learned from the operation of the existing

[1] The capital cost of a project if it could be constructed overnight. This cost does not include the cost of financing, the escalation due to increased material and labor costs, and the cost of inflation during construction.

fleet of nuclear power plants. The compilation of these 'user requirements' has had a tremendous impact in achieving international consensus regarding commonly acceptable safety requirements and performance expectations that would facilitate development of standardized designs which can be built in many countries without requiring significant redesign efforts.

Shortening the duration of the plant construction is important because of the interest and financing charges that are accrued during this period without countervailing revenue. However, it is important to remember that the objective is to reduce the overall plant cost, which means an optimization of construction schedule, construction costs and construction quality. It would not be meaningful to reduce the overall schedule period if that would result in an increase of the overall spending or incur later maintenance costs in a way that negates the savings in interest during construction. Recent nuclear construction projects have achieved optimum construction duration, cost and quality by streamlining the construction methods, using advanced construction technologies (all-weather construction, slip forming, open-top construction, modularization and prefabrication, advanced concrete mixing and pouring, automatic welding) and effective project management practices (IAEA, 2002) Figures 9.1 and 9.2 illustrate two recent examples of advanced construction projects. The use of effective procurement and contracting, as well as the close coordination with all relevant regulatory authorities, are also important contributors to this optimization.

One of the innovations incorporated in the design of evolutionary nuclear reactors is the use of modularization and factory prefabrication for both structural and system modules. Modules are fabricated in a controlled

9.1 Advanced concrete mixing and pouring at Sanmen 1 (China).

9.2 Modular construction at Lingao (China), the containment dome was assembled on the ground at the site and installed as a single module (weight 143 tons, diameter 37 m, height 11 m).

environment in a factory or in a workshop on the plant site, which normally improves their quality as compared to the traditional on-site stick build technique. Multiple modules can be fabricated in factories or workshops, while the civil work is progressing on the site in preparation to receive the modules. On the site, only sequential assembly of the modularized assemblies is required. This reduces on-site congestion, improves accessibility for personnel and materials, and can improve the construction schedule. It can also significantly reduce the manpower needs for the construction site work.

Another way to reduce the overall capital costs involves taking advantage of *economies of scale* and thus designing larger reactors where the capital costs can be amortized faster due to the larger electricity production. On the other hand, for some market conditions, increasing plant size to capture economies of scale may result in plants too large for many national grids or to meet the incremental demand. This has resulted in recent times in a parallel trend that encourages the development of small or medium-sized reactors (SMR) that are more affordable and can be built in a phased manner up to the total desired power based on the electricity demand or the financial means of the owner. These smaller designs may also be ideal for newcomer countries with small electric grids and/or limited financial resources. SMRs have the potential to capture *economies of series production* instead of economies of scale, if several units are constructed.

Most evolutionary designs also include design features that allow for plant lifetimes of 60 years and longer, thus spreading the capital investment over a longer plant operation life.

Evolutionary reactor designs have also been designed to achieve lower operating costs, such as with the optimization of the fuel cycle that brings savings in the form of increased plant availability, more effective use of fissionable resources, and minimization of waste and used fuel quantities and management costs. These designs also strive to obtain higher thermal efficiencies by using high-performance advanced turbines and sophisticated thermodynamic cycles, as well as expanded non-electrical applications. At the same time, since these new designs are expected to operate under higher demands, they employ improved corrosion-resistant materials and take advantage of major advances in fracture mechanics and non-destructive testing and inspection.

Evolutionary reactor designs employ several means to obtain performance improvement, such as the use of highly reliable 'smart' components (instrumented and monitored) able to detect incipient failures and to monitor their own performance. The effective application of smart components allows reducing the dependence on costly redundancy and diversity practices and permits the optimization of maintenance and replacement schedules. Most evolutionary designs also incorporate in-service testing and maintenance, thus further improving capacity factors.

The design, operation and maintenance of evolutionary reactors show an increased reliance on probabilistic risk assessment methods and databases that allow designers and operators to focus their efforts on the systems and components with higher risk of failure. The use of advanced computer modeling and simulation tools have fostered significant improvements in plant design and layout, plant arrangement and system accessibility, and in design features that facilitate decommissioning. In fact, current 'multidimensional' project management software is able to manage and coordinate all aspects of the life of a plant from the design to the operation and maintenance, including configuration control and other key construction processes such as procurement, manufacture, inventory, spare parts, costs and schedules. Another area that has had significant impact in the elimination of over-design and excessive safety margins has been the improvement of the technology base (i.e. improved understanding of thermo-hydraulic phenomena, more accurate databases of thermo-hydraulic relationships and thermo-physical properties, better neutronic and thermo-hydraulic codes, and further code validation). The only margins still accounted for in the design are simply associated with the limitations of calculation methodologies and uncertain data.

One of the best-known improvements incorporated into many evolutionary reactors is the use of passive safety systems that utilize gravity, natural convection and temperature and pressure differentials, enabling these systems to function without electrical power supply and/or actuation by powered instrumentation and control systems.

Advanced nuclear reactors have also paid increased attention to the effect of internal and external hazards in the design, in particular the seismic design and the qualification of buildings. At the same time, many of these designs have placed increased emphasis on the prevention and mitigation of severe accidents.

Last, but not least, advanced designs have taken advantage of the rapid progress in the field of control and instrumentation, in particular, with the introduction of microprocessors into the reactor protection system and with the use of digital instrumentation and control (I&C). An important development in these designs is the increased emphasis on the human–machine interface, including improved control room design and plant design for ease of maintenance.

9.4 Advanced nuclear reactor designs

This section provides descriptions of the technology options currently available for newcomer countries, in particular evolutionary reactor designs, as these are the most likely candidate technologies for most countries' first nuclear power plant, particularly in the near to middle term. For completeness, however, a brief discussion examining future trends for the development of nuclear reactors in the long term has also been included.

9.4.1 Evolutionary reactor designs

As described above, evolutionary designs achieve improvements over existing designs through small to moderate modifications, with a strong emphasis on maintaining design proveness to minimize technological risks. Not surprisingly, most of these are water-cooled reactors, as this type of design is the one where the nuclear community has more lessons learned and expertise.

The following designs, which have been ordered alphabetically herein, are those in a more advanced stage of development and would presumably be available for near-term deployment. In some cases, they have even been built or are in the process of being built somewhere in the world, and this will be indicated. The detailed technical data for all these designs can be found in IAEA (2010).

ABWR

The Advanced Boiling Water Reactor (ABWR), which is available from two competing vendors (GE-Hitachi and Toshiba, Fig. 9.3), combines the

9.3 The ABWR design (Toshiba).

best BWR design features from Europe, Japan and USA. The ABWR was developed in direct response to the EPRI Utility Requirements Document (URD) (EPRI, 1995, 1999), it is licensed in the USA, Japan and Taiwan (China) and it is the first evolutionary reactor design to operate commercially. There are currently four ABWRs in operation in Japan (Kashiwazaki-Kariwa 6 and 7, Hamaoka-5 and Shika-2), two in construction in Taiwan, China, and several more planned in Japan and the USA. In this sense, there is a proven capital and operation and maintenance cost structure associated with this design. The ABWR was designed with a shorter construction schedule in mind, by taking advantage of existing prefabricated construction experience and applying it into a modularized design. Although existing ABWRs are 1370 MWe, future ones are expected to be 1500 MWe as the reactor core has enough margins for these uprates. The ABWR has fully digital I&C and has adopted reactor internal pumps that eliminate the need for the large external recirculation coolant loops that involved penetrations below the top of the core elevation, thus making it possible to maintain core coverage during a postulated loss-of-coolant accident. This design also includes the capability to mitigate severe accidents and to reduce off-site consequences of accidents. The ABWR containment vessel is made of reinforced concrete with an internal steel liner.

ABWR-II

The Advanced Boiling Water Reactor II (ABWR-II), developed by GE-Hitachi, is a further enhancement of the ABWR. It offers a larger power output of up to 1700 MWe, due to a larger core with 1.5 times larger fuel bundles and the control rods arranged in a K-lattice (as opposed to the conventional N-lattice[2]). This new core design may also provide increased flexibility for higher burn-up, use of MOX fuel and higher conversion rate configurations. The ABWR-II also includes a modified Emergency Core Cooling System, and an optimum combination of active and passive heat removal systems, resulting in a design that promises better economics, performance and safety.

ACR-1000

The Advanced CANDU Reactor-1000 (ACR-1000) design is a 1200 MWe pressure tube reactor that retains many essential features of a typical CANDU plant design, including horizontal fuel channel core, a low-temperature heavy water moderator, a water-filled reactor vault, two independent safety shutdown systems, a highly automated control system, on-power fueling and a reactor building that is accessible for on-power maintenance and testing. The key differences from the traditional CANDU design incorporated into the ACR-1000 are the use of low-enriched uranium fuel (as opposed to natural uranium), the use of light water instead of heavy water as the reactor coolant, and a lower moderator volume to fuel ratio. These features together with a number of other evolutionary changes lead to the many benefits for the ACR-1000 design: a more compact core design, an increased burn-up as a result of the fuel enrichment, increased safety margins, improved overall turbine cycle efficiency through the use of higher pressures and higher temperatures in the coolant and steam supply systems, reduced emissions through the elimination of tritium production in the coolant and other environmental protection improvements, enhanced severe accident management by providing backup heat sinks, improved performance through the use of advanced operational and maintenance information systems, and improved separation of redundant structures, systems and components (SSCs) important to safety through the use of a four-quadrant plant layout. The ACR-1000 design has been reviewed by the Canadian regulatory body and has been given a positive regulatory opinion about its licensability. The generic preliminary safety analysis report for the ACR-1000 design was completed in September 2009. The final stage of the ACR-1000 design is currently underway including documentation and additional confirmatory analysis, and the basic engineering is expected to be completed in 2010.

[2] In a K-lattice there are two control rods for every four fuel bundles, while in the traditional N-lattice there is one control rod for every four fuel bundles.

AP1000

The Westinghouse Advanced Passive PWR (AP1000) is a two-loop 1117 MWe PWR scaled up from that already certified in the USA AP600 design, which was originally compliant with the EPRI URD (EPRI, 1995, 1999). In the AP1000, designers have made an effort to simplify all systems, and to reduce the number of systems and components for easier construction, operation and maintenance. As in other evolutionary concepts, the AP1000 uses prefabrication and modular construction as a way to reduce construction schedule uncertainties. One of the signature characteristics of the AP1000 is the use of passive safety systems, i.e., those that rely on natural driving forces such as pressurized gas, gravity flow, natural circulation flow, and convection, for core cooling, containment isolation, residual heat removal and containment cooling. On the other hand, the plant design utilizes proven technology and capitalizes on more than 40 years of PWR operating experience. The AP1000 also incorporates severe accident mitigation features, such as in-vessel retention of core debris following a core melt event, and no reactor vessel penetrations below the top of the core level. Two AP1000 projects are currently under construction in China (Haiyang and Sanmen) and substantial construction and operating experience is expected from these. In the USA, final design certification by the US NRC for the AP1000 is expected by 2011 and there are several applications for its construction starting 2011.

APR1400

The Advanced Power Reactor 1400 (APR1400), with a rated power of 1400 MWe, is the largest two-loop PWR currently available. The APR1400 is an evolutionary reactor developed in the Republic of Korea, based on the accumulated experience from the design and operation of the 1000 MWe Korean Standard Nuclear Power Plant and from the EPRI URD (EPRI, 1995, 1999). The APR1400 incorporates a number of improvements to meet operators' needs for enhanced safety, performance and economics and to address new licensing requirements such as the mitigation of severe accidents. The APR1400 has a very characteristic configuration, with two large steam generators and four reactor coolant pumps in a 'two hot legs and four cold legs' arrangement. The APR1400 also features fully digital instrumentation and control (I&C), and a main control room designed with full consideration of human factors. The APR1400 incorporates safety systems with both active and passive characteristics, and has also been designed to take advantage of modularization and prefabrication construction techniques to ensure a predictable construction budget and schedule. Two

APR1400 units are currently under construction in the Republic of Korea (Shin-Kori 3 and 4), and they are expected to enter commercial operation in 2013–14. The APR1400 has also been selected for the first four units that will be built in the United Arab Emirates.

APWR

The Advanced Pressurized Water Reactor (APWR) is a four-loop PWR developed jointly by a group of Japanese utilities, Mitsubishi Heavy Industries (MHI) and Westinghouse, that relies on a combination of active and passive safety systems. It is currently made available by MHI. The high-capacity APWR, with 1534 MWe (1700 MWe in Europe and the US), takes advantage of economies of scale and uses high-performance steam generators and low-pressure turbines with very large last-stage blades. The APWR allows operation with long fuel cycles, and increased flexibility such as the use of low-enriched fuel in order to reduce uranium requirements, the use of MOX cores and high burn-up fuels. The neutron economy and the long-term reliability of the reactor vessel have been improved with the use of a neutron reflector. The container includes a steel liner to prevent leakage surrounded by the concrete structure that provides structural protection. As in other evolutionary designs, the construction of the APWR also takes advantage of modularization and advanced design, simulation and management computer programs. Two APWRs are planned to be built at Tsuruga-3 and 4 in Japan, and several more in the United States.

EC6

The Enhanced CANDU 6 (EC6) is a 740 MWe pressure tube reactor designed by Atomic Energy of Canada Limited (AECL). The EC6 design benefits from the proven principles and characteristics of the CANDU 6 design, which is currently in operation in several countries in the world, such as natural uranium fuel, two independent safety shutdown systems, a separate low-temperature, low-pressure moderator (which provides an inherently passive heat sink by permitting heat to be removed from the reactor core under abnormal conditions), a reactor vault that is filled with cool light water (which surrounds the reactor core, providing another passive heat sink), on-power refueling, and a modular, horizontal fuel channel core. The EC6 design includes a more robust containment with thicker walls and a steel liner, enhanced severe accident management, addition of the emergency heat removal system as a safety system, improved shutdown performance for larger loss of coolant accident margins, and a plant life of 60 years with one life extension of critical equipment such as fuel channels and feeders at mid-life. The Canadian Nuclear Safety Commission (CNSC) is currently conducting the design review of the EC6.

EPR

The European Pressurized Water Reactor (EPR) is the result of a joint development effort by Framatome and Siemens, and now made available by AREVA. The EPR is a very robust 1600+ MWe four-loop PWR design, with a small technology leap. In the EPR, the designers have chosen to use active safety systems and increase the redundancy in the power sources and the water inventories to manage any potential transients. The EPR also has a double concrete containment and a core catcher for the mitigation of severe accidents. The core of the EPR is designed to operate with both UO_2 and MOX fuel, and is expected to provide reduced uranium consumption. The EPR has been designed to operate under load following conditions between 20% and 100% of rated generator power. The EPR includes fully digital I&C systems, but does not take advantage of modular construction and factory fabrication. EPR reactors are currently under construction in Finland, France and China, and planned in the US and India.

ESBWR

GE Hitachi Nuclear Energy's Economic Simplified Boiling Water Reactor (ESBWR) is a 1520 MWe power plant design based on the earlier 670 MWe Simplified Boiling Water Reactor (SBWR) design. The ESBWR design incorporates innovative, yet proven, features to further simplify an inherently simple direct cycle nuclear plant. The ESBWR completely relies on passive safety systems for both normal and off-normal operating conditions, such as natural circulation, isolation condensers or gravity-driven cooling systems. The core of the ESBWR is shorter and the overall vessel height is larger than a conventional BWR, in an effort to maximize natural circulation and avoid the use of recirculation pumps or their associated piping. The US NRC provided the ESBWR with an advanced Safety Evaluation Report (SER) with no open items in August 2010, and the final design certification is expected by September 2011.

KLT-40S

The KLT-40S is a pressurized water reactor based on the commercial KLT-40 marine propulsion plant, and is an advanced variant of the reactor plants that power nuclear icebreakers. The construction of a small-size floating nuclear cogeneration plant with two KLT-40S reactors is currently under way in Russia. The KLT-40S is a modular design in which the reactor, steam generators and main circulation pumps are connected with short nozzles. It is a four-loop system including forced and natural circulation of the primary coolant, with a once-through coiled steam generator, an external gas pressurizer system and passive safety systems.

PHWR

India has developed its own indigenous Pressurized Heavy Water Reactor (PHWR) design that consists of 220 MWe, 540 MWe and 700 MWe units. India is currently operating 16 units of 220 MWe and two units of 540 MWe. Construction of two 700 MWe units is underway. The Indian PHWR was developed from the experience in the operation of earlier units and from indigenous R&D efforts. The important features introduced in these units include two diverse and fast-acting shutdown systems, double containment of the reactor building, water-filled calandria vault, integral calandria end shield assembly, and calandria tube filled and purged with carbon dioxide to monitor pressure tube leak by monitoring the dew point of carbon dioxide. These units also include a valve-less primary heat transport system and a simplified control room concept, as well as advanced control and instrumentation systems that incorporate computer-based systems to match with the advancement in technology.

WWER

The WWER-1000 is a four-loop pressurized water-cooled reactor that incorporates active and passive safety systems, and partly adapts to Western standards the substantial design and operating experience accumulated in the Russian Federation in the last 50 years. It is currently operating in the Russian Federation and under construction in China, India and Iran. The WWER-1200 is a scaled-up version of the WWER-1000. Like its predecessor, it is a four-loop design with horizontal steam generators, which have a track record of providing the longest operating life. The WWER-1200 also includes active and passive safety systems, a double containment and severe accident mitigation systems, such as a core catcher. Both the WWER-1000 and WWER-1200 cores use the characteristic hexagonal fuel assemblies (as opposed to the traditional square ones used by all other water-cooled reactor designs) and would allow for the possibility of using MOX fuel. It is currently under construction in the Russian Federation and planned for construction in Bulgaria. Since there are some WWER-1000 and WWER-1200 units currently in operation or under construction, this design has a proven construction schedule, as well as some operating experience.

The following designs, also ordered alphabetically, are also evolutionary concepts but their current stage of development indicates that they would only be available for deployment in the middle-term. The detailed technical data for all these designs can be found in ARIS (IAEA, 2010).

AHWR

The Indian Advanced Heavy Water Reactor (AHWR) has been designed by Bhabha Atomic Research Center (BARC) to achieve large-scale use of thorium for the generation of commercial nuclear power. This reactor will produce most of its power from thorium, with no external input of uranium-233 in the equilibrium cycle. The AHWR is a 300 MWe, vertical, pressure tube type, boiling light water-cooled, and heavy water-moderated reactor. The reactor incorporates a number of passive safety features and is associated with a closed fuel cycle, thus having reduced environmental impact. At the same time, efforts have been made to incorporate several features that are likely to reduce its capital and operating costs. The basic design of the reactor and detailed design of its major nuclear systems have been completed. The research, design, and demonstration (RD&D) for AHWR has been and is being performed at the BARC. The Indian Atomic Energy Regulatory Board (AERB) has carried out a pre-licensing safety appraisal of the AHWR. Subsequently, the regulatory clearances for different stages of construction, starting from plant siting and procurement of long-delivery major equipment, will be progressively sought. The construction of the AHWR prototype is likely to commence in 2011.

ATMEA1

ATMEA1 brings together field-proven technology that is already incorporated into AREVA's EPR and MHI's APWR. It is a three-loop PWR that relies primarily on active safety systems, and incorporates severe accident mitigation features. Fuel cycle lengths can be set to be from 12 to 24 months. Fuel management variations in ATMEA1 can go from a full uranium oxide core to a mixed core with MOX fuel up to one-third of the core for the standard design, and up to 100% without any major design modification. The core design includes a radial neutron reflector that improves neutron utilization, thus reducing the fuel consumption, and reduces the irradiation to the vessel.

CAREM

CAREM (in Spanish, Central Argentina de Elementos Modulares) is an Argentinian nuclear reactor that has an indirect-cycle reactor with some distinctive and characteristic features that greatly simplify the design, such as an integrated primary cooling system, self-pressurized primary system and safety systems relying on passive features. The first step of this project is the construction of a 27 MWe (CAREM-25) prototype in Argentina. CAREM has been recognized as an International Near Term Deployment (INTD) reactor by the Generation IV International Forum (GIF).

IMR

The Integrated Modular Water Reactor (IMR) is a medium-sized power reactor with a reference output of 350 MWe and an integral primary system reactor with potential deployment after 2020 developed by Mitsubishi Heavy Industries in Japan. IMR employs the hybrid heat transport system, which is a natural circulation system under bubbly flow condition for primary heat transportation, and no penetrations in the primary cooling system by adopting the in-vessel control rod drive mechanism. These design features allow the elimination of the emergency core cooling system. Because of its modular characteristics, it is suitable for large-scale power stations consisting of several modules and also suitable for small power stations, especially when the capacity of the grid is small. IMR also has the capability for district heating, seawater desalination, process steam production, and so forth.

IRIS

IRIS is a modular light water reactor with an integral primary system configuration designed by an international group of 20 organizations from nine countries led by Westinghouse. ARIS has a simplified compact design where the primary vessel houses steam generators, pressurizer and pumps; a novel safety approach; and an optimized refueling cycle with intervals of at least four years. Due to its integral configuration, in IRIS a variety of accidents are by design either eliminated or their consequences and/or probability of occurring are greatly reduced. This provides a superb defense in depth which may allow IRIS to support a claim of no need for an emergency response zone.

KERENA

KERENA (earlier known as SWR-1000) is an evolutionary boiling water reactor based on the experience gained from the proven engineering of current generation BWR plants supplemented by an innovative approach. The current final basic design of KERENA is part of a strategic partnership between AREVA and the German utility E.ON Kernkraft. In KERENA, safety systems have been simplified by introducing passive safety systems, and most nuclear safety functions are performed by active systems with a passive system as backup. The core height has been reduced to promote natural circulation, and the eight reactor water recirculation pumps are the so-called wet-motor pumps, where the electric pump motor is situated inside the reactor coolant pressure boundary.

mPower

The mPower, designed by Babcock and Wilcox (B&W) in the USA, is a scalable and modular system in which the nuclear core and steam generators are contained within a single vessel. It is a modular reactor designed to match customer demand in 125 MWe increments. mPower employs an integral nuclear system design, passive safety systems, a 4.5-year operating cycle between refueling, 5% enriched fuel, secure underground containment, and spent fuel pool capacity for the life of the plant. A scaled prototype of mPower using electric heating instead of nuclear heating is currently under construction in the USA to verify the reactor design and safety performance, supporting its licensing activities with the US Nuclear Regulatory Commission.

NuScale

NuScale is a reactor design based on a concept originally proposed by Oregon State University in the USA. A NuScale plant can consists of one to 12 independent modules, each capable of producing a net electric power of 45 MWe. Each module includes a Pressurized Light Water Reactor operated under natural circulation primary flow conditions. Each reactor is housed within its own high-pressure containment vessel which is submerged underwater in a stainless steel-lined concrete pool. In early 2008, NuScale Power notified the US Nuclear Regulatory Commission of its intent to begin pre-application discussions aimed at submitting an application for design certification of a 12-module NuScale power plant.

SMART

SMART is a 300 MWt/100 MWe integral-type PWR designed by KAERI in the Republic of Korea. SMART incorporates inherent safety features such as the integral configuration of the reactor coolant system, its improved natural circulation capability, a passive residual heat removal system and an advanced LOCA mitigation system. SMART has a low power density core that uses 5 w/o UO_2 and results in a thermal margin of more than 15% to accommodate any design basis transients with regard to the critical heat flux. SMART has been conceived as a multipurpose energy source including non-electric applications such as seawater desalination, district heating or other industrial applications.

9.4.2 Innovative reactor designs

As discussed in Section 9.2, evolutionary reactor designs have concentrated on improving the economics and the performance of existing nuclear reactors. Beyond 2030, it is anticipated that new reactor designs will address key issues such as the closure of the fuel cycle or proliferation concerns while possibly ensuring competitive economics, safety and performance.

In order for nuclear energy to remain a long-term option in the world's energy mix, nuclear power technology development must meet sustainability goals with regard to natural resource utilization and radioactive waste management. Interest in fast neutron systems and the related fuel cycles has reappeared with the realization of their requisite role in meeting these goals.

Also in alignment with the above sustainability goals, there has been an increased interest in expanding the range of energy products provided by nuclear fission beyond electricity production to include industrial heat, hydrogen and energy for transportation. In this sense, interest in high-temperature gas-cooled reactors that may be able to realize these applications in the most effective manner has also intensified.

Both fast reactors and high-temperature gas-cooled reactors, among others, are innovative reactor designs and the achievement of their full potential is conditional on continued advances in research and technology development.

A few of the topics in which significant work is presently taking place are briefly presented below.

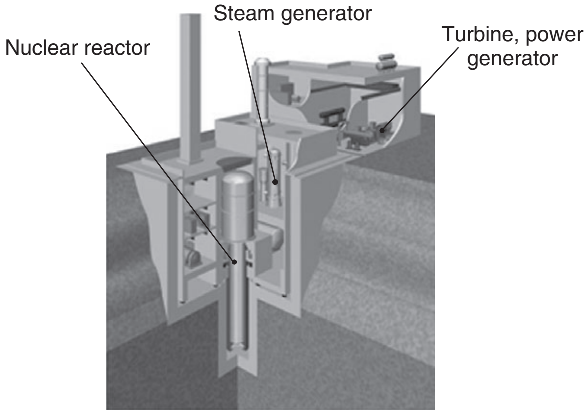
- *Core physics: neutronics and thermal-hydraulics*. Improvements to safety and economics are met through stringently accurate design requirements, which must be demonstrated with well-validated calculation tools. Presently, in the area of neutronics, the uncertainties on the nuclear data relevant to several of these innovative reactor designs in many cases are such that they negate the benefits offered by advanced modeling and simulation techniques. In the area of thermal-hydraulics, the extreme thermal performance expected from these systems imposes the need for the accurate determination of thermal-hydraulic parameters to relatively high resolution in order to ensure that the relevant safety criteria are met in both normal and off-normal operation.
- *Fuels and structural materials.* Oxide-, metallic-, carbide- and nitride-based fuels which incorporate depleted, natural and recycled uranium as well as possibly recycled plutonium, and even thorium, are considered for use in innovative reactors. Advanced structural materials with increased strength and creep resistance are sought as a means by which to improve reactor performance by allowing for greater design and safety margins, longer lifetimes and higher operating temperatures; in

parallel, this leads to improved economics through reduction of plant materials and permitted flexibility for design simplifications.

- *Coolants and coolant technologies.* The coolants under consideration for innovative reactors include alkali metals (e.g. sodium), heavy liquid metals (e.g. lead and lead–bismuth eutectic) and gases (e.g. helium and supercritical water). Each demonstrates distinct advantages and disadvantages with regard to performance and safety, so much so that serious studies are devoted to all.

Several suppliers and designer organizations are working on the development of innovative nuclear reactor concepts that address the above concerns, although most of them are still in an early stage of development. Some of these are as follows.

- *4S*. Toshiba's 4S (super-safe, small and simple; see Fig. 9.4) is a small sodium-cooled reactor without on-site refueling in which the core has a lifetime of approximately 30 years. Being developed as a distributed energy source for multipurpose applications, the 4S offers two outputs of 30 MWt and 135 MWt, respectively selected from demand analyses. Although 4S has a fast neutron spectrum it is not a breeder reactor since blanket fuel, usually consisting of depleted uranium located around the core to absorb leakage neutrons from the core to achieve breeding of fissile materials, is not provided in its basic design. The reactor power can be controlled by the water/steam system without affecting the core operation directly. The capability of power self-adjustment makes the reactor applicable for a load-follow operation mode. The reactor is a pool type, integral type as all primary components are installed inside the reactor vessel.



9.4 The 4S Design (Toshiba).

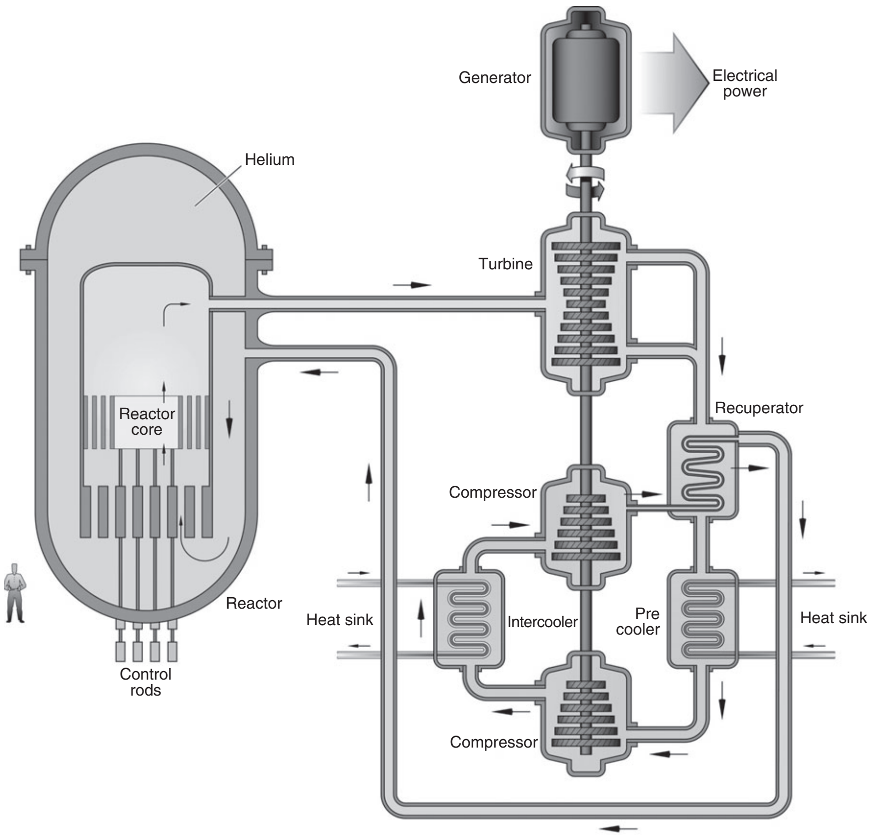
- *PRISM*. The PRISM reactor is a 2200 MWe modular sodium-cooled fast reactor whose development began as a joint project between General Electric (GE) and the US Department of Energy as part of the advanced liquid-metal reactor (ALMR) program. Development has since continued at GE-Hitachi, and the design today incorporates Generation IV objectives. PRISM employs passive safety and uses a modular fabrication technique to expedite plant construction. PRISM uses metallic fuel for better compatibility with the coolant, inherent safety, and ease of fabrication in a hot cell.

Beginning in 2000, 10 countries joined together in the Generation IV International Forum (GIF) to perform the necessary research and development to support the development and deployment of innovative nuclear energy systems that can be licensed, constructed, and operated in a manner that will provide competitively priced and reliable energy products while satisfactorily addressing nuclear safety, waste, proliferation, and public perception concerns (GIF, 2002). GIF experts assessed a large number of candidate reactor concepts by using the common GIF evaluation methodology resulting in the following six reactor systems selected.

The *Gas-Cooled Fast Reactor* (GFR, Fig. 9.5) system features a fast-neutron spectrum and closed fuel cycle for efficient conversion of fertile uranium and management of actinides. The reference reactor is a 600-MWth/288-MWe, helium-cooled system operating with an outlet temperature of 850°C using a direct Brayton cycle gas turbine for high thermal efficiency. Several fuel forms are being considered for their potential to operate at very high temperatures and to ensure an excellent retention of fission products: composite ceramic fuel, advanced fuel particles, or ceramic clad elements of actinide compounds. Core configurations are being considered based on pin- or plate-based fuel assemblies or prismatic blocks. The GFR is estimated to be deployable by 2025.

The *Lead-Cooled Fast Reactor* (LFR, Fig. 9.6) system features a fast-neutron spectrum and a closed fuel cycle for efficient conversion of fertile uranium and management of actinides. The system uses a lead or lead/bismuth eutectic liquid-metal cooled reactor. Several options with different plant sizes have been proposed including a battery of 50–150 MWe that features a very long refueling interval, a modular system rated at 300–400 MWe, and a large monolithic plant option at 1200 MWe. The fuel is metal or nitride-based, containing fertile uranium and transuranics. The LFR system is estimated to be deployable by 2025.

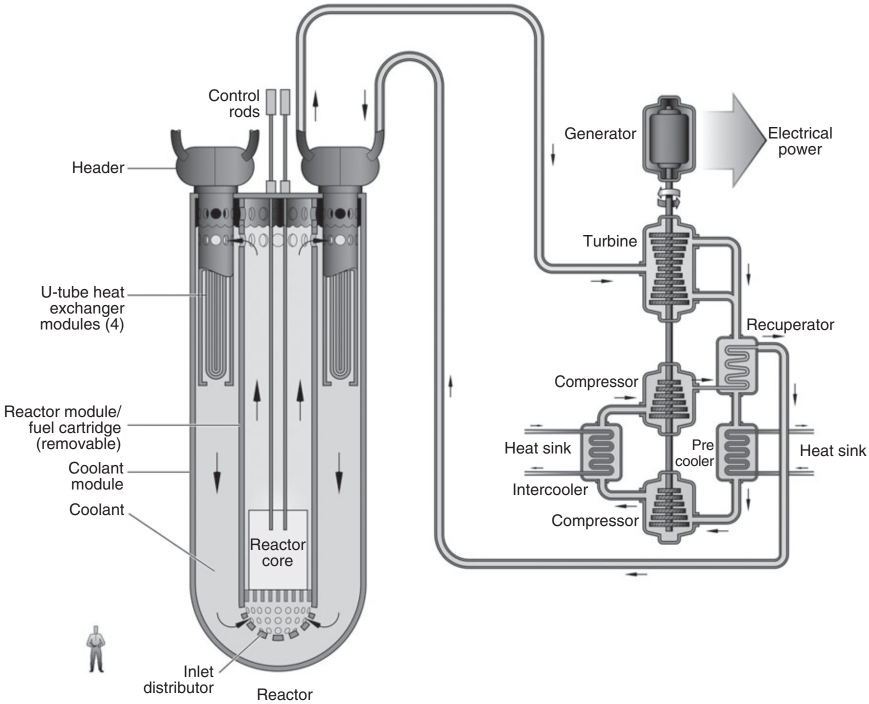
The *Molten Salt Reactor* (MSR, Fig. 9.7) system features an epithermal to thermal neutron spectrum and a closed fuel cycle tailored to the efficient utilization of plutonium and minor actinides. In the MSR system, the fuel is a circulating liquid mixture of sodium, zirconium, and uranium fluorides.



9.5 GIF Gas-cooled Fast Reactor (GFR) generic concept (illustration courtesy of Idaho National Laboratory).

The molten salt fuel flows through graphite core channels, producing a thermal spectrum. The heat generated in the molten salt is transferred to a secondary coolant system through an intermediate heat exchanger, and then through another heat exchanger to the power conversion system. There is no need for fuel fabrication. The reference plant has a power level of 1000 MWe. The system operates at low pressure (<0.5 MPa) and has a coolant outlet temperature above 700°C, affording improved thermal efficiency. The MSR is estimated to be deployable by 2025.

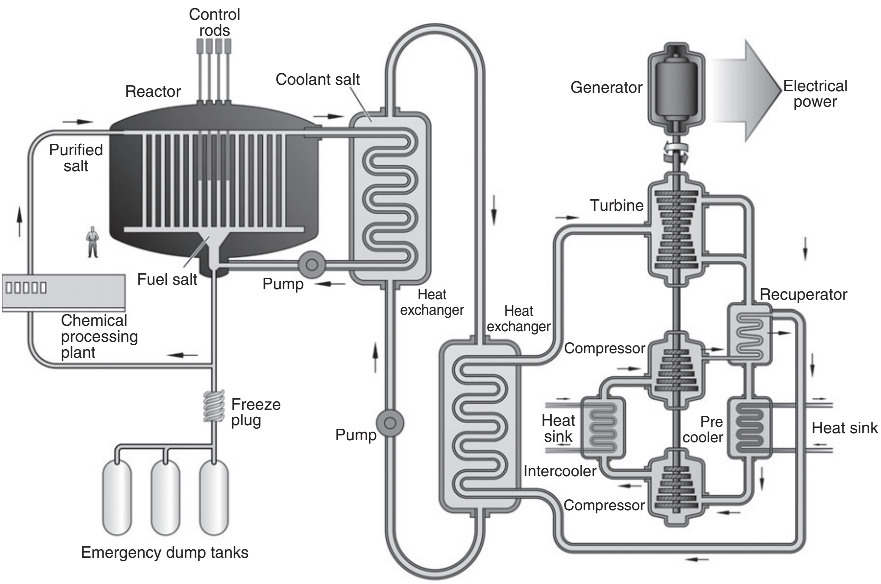
The *Sodium-Cooled Fast Reactor* (SFR, Fig. 9.8) system features a fast-neutron spectrum and a closed fuel cycle for efficient conversion of fertile uranium and management of actinides. Two options have been envisioned. The first is an intermediate-size (150 to 500 MWe) sodium-cooled reactor with a uranium–plutonium–minor-actinide–zirconium metal alloy fuel,



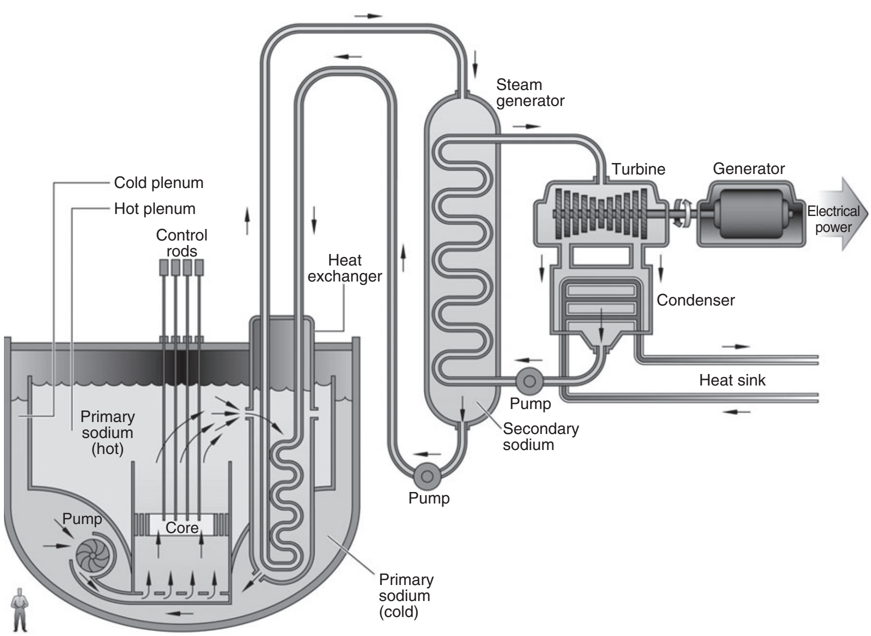
9.6 GIF Lead-cooled Fast Reactor (LFR) generic concept (illustration courtesy of Idaho National Laboratory).

supported by a fuel cycle based on pyrometallurgical processing in colocated facilities. The second option is a medium to large (500 to 1500 MWe) sodium-cooled fast reactor with mixed uranium–plutonium oxide fuel, supported by a fuel cycle based upon advanced aqueous processing at a central location serving a number of reactors. The outlet temperature is approximately 550°C for both. The SFR is estimated to be deployable by 2015.

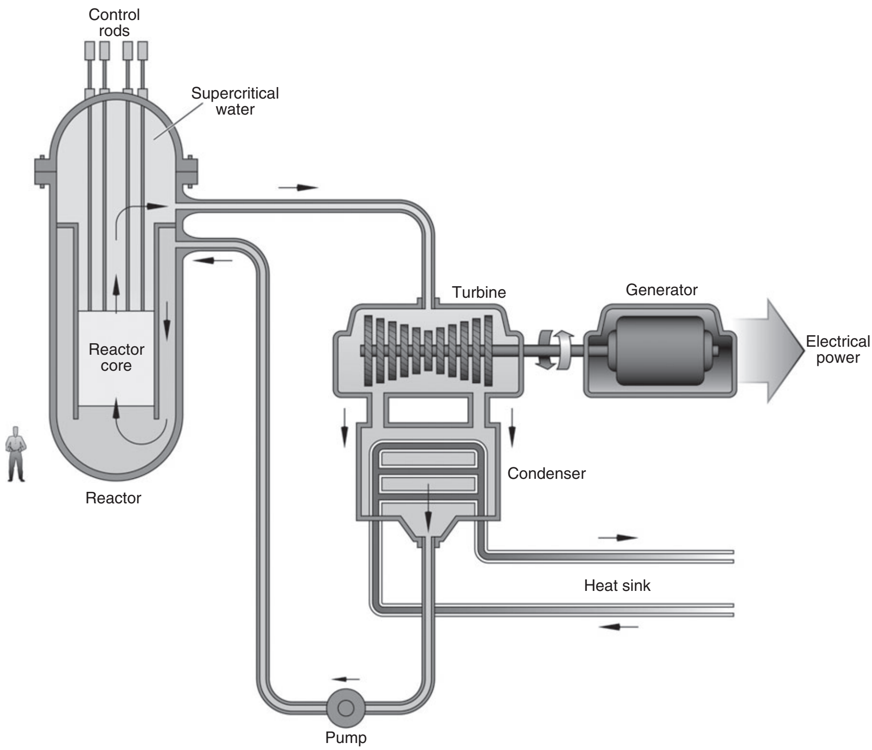
The *Supercritical-Water-Cooled Reactor* (SCWR, Fig. 9.9) system features two fuel cycle options: the first is an open cycle with a thermal neutron spectrum reactor; the second is a closed cycle with a fast-neutron spectrum reactor and full actinide recycle. Both options use a high-temperature, high-pressure, water-cooled reactor that operates above the thermodynamic critical point of water (22.1 MPa, 374°C) to achieve a thermal efficiency approaching 44%. In either option, the reference plant has a 1700-MWe power level, an operating pressure of 25 MPa, and a reactor outlet temperature of 550°C. Passive safety features similar to those of the simplified boiling water reactor are incorporated. The balance-of-plant is considerably



9.7 GIF Molten Salt Reactor (MSR) generic concept (illustration courtesy of Idaho National Laboratory).



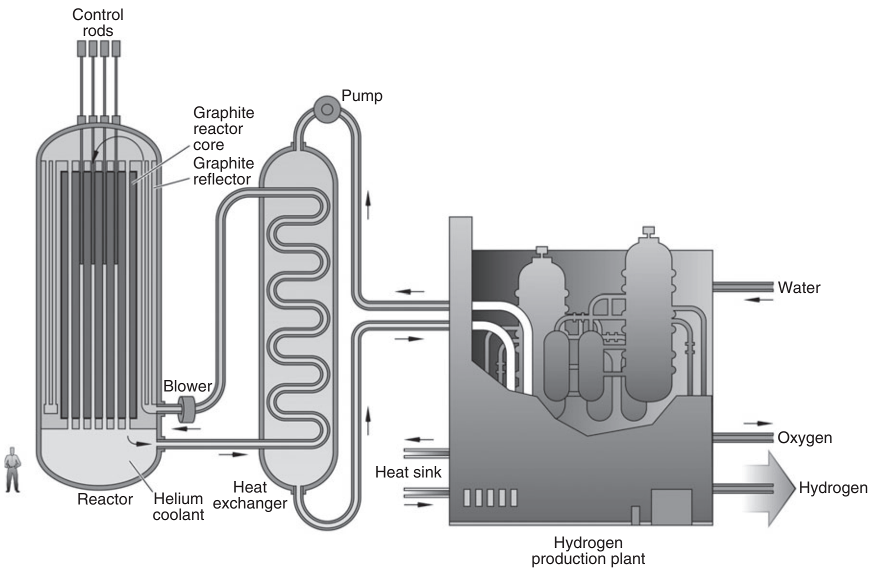
9.8 GIF Sodium-cooled Fast Reactor (SFR) generic concept (illustration courtesy of Idaho National Laboratory).



9.9 GIF Supercritical Water Cooled Reactor (SCWR) generic concept (illustration courtesy of Idaho National Laboratory).

simplified because the coolant does not change phase in the reactor. The SCWR system is estimated to be deployable by 2025.

The *Very-High-Temperature Reactor* (VHTR, Fig. 9.10) system uses a thermal neutron spectrum and a once-through uranium cycle. The VHTR system is primarily aimed at relatively faster deployment of a system for high-temperature process heat applications with superior efficiency (see Section 9.5). The reference reactor concept has a 600-MWth helium-cooled core based on either the prismatic block fuel of the Gas Turbine–Modular Helium Reactor (GT-MHR) or the pebble fuel of the Pebble Bed Modular Reactor (PBMR). The primary circuit is connected to a steam reformer/steam generator to deliver process heat. The VHTR system has coolant outlet temperatures above 1000°C. The system may incorporate electricity generation equipment to meet cogeneration needs. The system also has the flexibility to adopt U/Pu fuel cycles and to offer enhanced waste minimization. The VHTR system is the nearest-term hydrogen production system, estimated to be deployable by 2020.

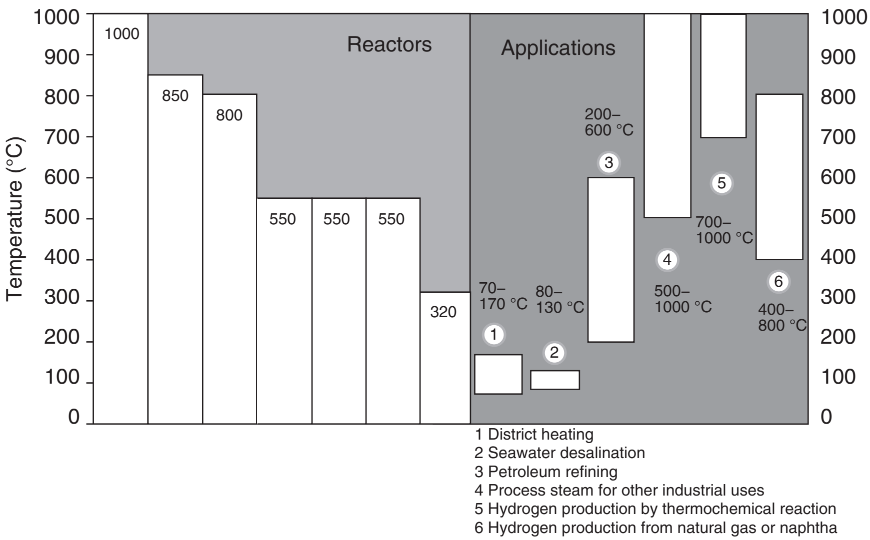


9.10 GIF Very High Temperature Reactor (VHTR) generic concept (illustration courtesy of Idaho National Laboratory).

9.5 Non-electrical applications

When evaluating the various nuclear technology options available, it is important to keep in mind that nuclear power also has important potential in the area of non-electric applications such as desalination, hydrogen production, district heating, oil refining, tertiary oil recovery or coal gasification (see Fig. 9.11). Indeed, there is experience with nuclear power in the heat and steam market in the low-temperature range, i.e. desalination and district heating. An extension appears possible on a short term in these areas as well as for tertiary oil recovery. The petrochemical and refining industries represent another huge potential with their growing demand for hydrogen and process steam due to the increasing share of fossil fuels such as heavy oils, oil shale or tar sands entering the market. In the high-temperature heat market, nuclear is also applicable to the production processes of liquid fuels or of hydrogen by steam reforming or water splitting, compatible with the needs of the transportation sector. The feasibility of steam reforming of methane or coal gasification under nuclear conditions was already successfully demonstrated.

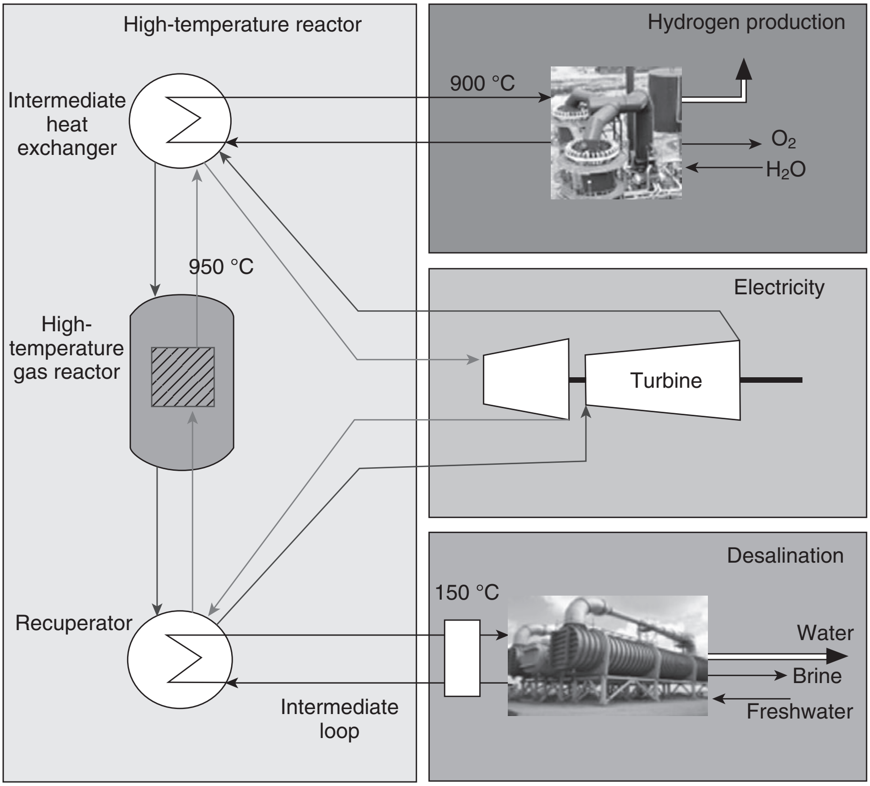
There are many other industrial sectors (such as paper and pulp, food industry, automobile industry, or textile manufacturing) which have a high demand for electricity and heat/steam at various levels of temperature and pressure. In such industrial processes, the reliability and availability of the



9.11 Potential non-electric applications of various nuclear reactor types.

energy supply is essential, demanding the continuous operation of their process units approaching 100%. The temperatures required cover a wide spectrum. With respect to the required capacity, 99% of the industrial users need thermal power of less than 300 MW, which accounts for about 80% of the total energy consumed. Half of the industrial users even demand thermal power in the range of less than 10 MW. Ensuring supply security by diversification of the primary energy carriers and, at the same time, limiting the effects of energy consumption on the environment will become more important goals in the future.

In principle, any type and size of nuclear reactor can be used as heat source for various processes and applications. No technical impediments to coupling nuclear reactors to such applications have so far been observed, although a number of safety-related studies of coupled systems may still be necessary. Different types of nuclear reactors provide a different range of coolant temperatures. The higher the temperature, the larger is the range of applications and products. Current Light Water Reactors (LWR) are characterized by maximum temperatures of less than 320°C, only allowing steam production at a lower quality. Hence, they are mainly used for electricity generation with occasional steam extraction. In Fast Breeder Reactors (FBR) the coolant reaches a higher temperature of around 500°C, while High-Temperature Gas-Cooled Reactors (HTGR) are able to provide steam up to a temperature of 950°C. It is an area where nuclear energy specifically from HTGRs could play a major role in future (Fig. 9.12).



9.12 Cogeneration using high-temperature gas-cooled reactor.

The main challenges at present are to combine nuclear power and non-electric applications into a single strategy and to establish the transition technologies from present industrial practice to emerging new resources in order to stabilize energy cost. The renewed interest in nuclear power production may lead to an increased role for nuclear energy in the area of non-electric applications, which currently are almost entirely dominated by fossil fuel energy sources. Among other advantages (including less environmental impact and high energy content of nuclear fuel) of the use of nuclear energy for non-electric applications is that nuclear reactors offer process heat at a wide spectrum of temperatures from some 200°C to 1000°C, which covers practically the whole range required for most non-electric applications, including waste heat which can be harnessed in some very low-temperature non-electric applications such as seawater desalination.

Cogeneration may become the most suitable option for non-electric applications. In this case the steam and electricity can be produced with a single nuclear plant. The cogeneration mode has several practical advantages: an increased plant thermal efficiency, the possibility of varying the heat supply according to demand and an easier implementation, as almost all nuclear reactors for electricity production can be adapted. Thus the first nuclear non-electric applications are likely to be of the cogeneration type. This has been confirmed by the experience with nuclear district heating and desalination.

9.5.1 Seawater desalination

Due to water scarcity, total contracted desalination capacity (from both seawater and brackish water) has almost tripled in the past decade, reaching a global online capacity of about 50 million m^3/d. Desalination has proven during the last 50 years its reliability to deliver large quantities of fresh water from the sea. Technological advances of the last decade have helped desalination to spread faster and to become a reliable way to supply water and consequently to promote sustainable development. Among the drivers for the growing interest in seawater desalination using nuclear energy are cheaper energy, less uncertainty on energy costs, higher load factor of the desalination plant, better load factor of the nuclear unit, utilization of the nuclear plant's unused land, and reduction of the desalination carbon footprint. The future requires effective integration of energy resources to produce power and desalinated water economically with proper consideration for the environment.

The principal desalination processes are based either on distillation or on membrane separation. The first group includes the widely applied commercial methods of Multi-Stage Flash Distillation (MSF) and Multiple Effect Distillation (MED). Still under development is Thermal Vapor Compression distillation (TVC) which is a promising process with a higher conversion ratio. The main characteristics of distillation processes are high energy cost, independence from feed water quality and simple technology with wide experience worldwide. The processes using membranes are characterized by having lower energy costs, dependent on the feed water quality, and simplicity. Major thermal energy in the range of 100–130°C is required to heat the feed water.

All existing designs of nuclear reactors could be used to provide electricity, low-temperature heat and/or combinations of both as required for desalination. Relevant experience with nuclear desalination is already available. The use of nuclear heat requires a close location of the nuclear plant to the desalination plant, while the use of electricity generated by nuclear energy for reverse osmosis (RO) does not differ from any other use of electricity in that the energy source may be located far from the customer, with electricity being provided through the electricity grid. It should be noted, however, that electricity taken directly from the plant is cheaper than the electricity from the grid and that a distant location would not allow the use of warm water from a condenser for the RO feed.

Limited experience exists with nuclear desalination since the 1960s from nine nuclear units in Japan and one in Kazakhstan. The latter was a BN-350 fast reactor which produced 135 MWe and 80,000 m^3/d of fresh water by MED over 27 years before it was removed from operation in 1999. In Japan, nuclear desalination is experienced in the form of having the desalination plants constructed on-site of the nuclear power plant with aim at supplying the required make-up cooling water to these nuclear power plants. Such desalination plants have in general small capacities of 1000–3000 m^3/d. In India, a combined MSF and RO hybrid system connected to twin 170 MWe pressurized heavy water reactors has been constructed and is, presently, in the commissioning phase. With capacities of 1800 m^3/d by RO and 4500 m^3/d by MSF, it will become the largest nuclear-based desalination plant in the world. Optimization of water desalination using nuclear reactors has been analysed, and studies are still under investigation in several countries.

New developments in nuclear desalination are numerous as many countries have consistently progressed almost simultaneously in three technical fields: the development of improved or new generation nuclear reactors, the improvements in desalination technologies and the adoption of many cost reduction strategies. An interesting feature of this development is that many countries, normally not considered as exporting countries, have begun to develop their own nuclear reactors. For example, Argentina is developing the CAREM reactor. China is pursuing the development of the dedicated heat only reactor NHR-200 providing relatively low-temperature heat for an MED process, with some electricity production to meet the local electricity needs. India is going along with a consistent evolutionary approach to develop its advanced PHWRs. The Republic of Korea continues with its program to develop the System-integrated Modular Advanced Reactor (SMART). South Africa is developing the PBMR which can be used for electricity generation, hydrogen production and desalination (although the project is currently frozen).

9.5.2 Hydrogen production

As an alternative path to the current fossil fuel economy, a hydrogen economy is envisaged in which hydrogen would play a major role in all sectors of the economy by replacing fossil fuels. Indeed, the hydrogen economy has received much renewed interest because of the new developments in HTGR technology. Nuclear-generated hydrogen has important potential advantages over other sources that will be considered for a growing hydrogen economy. Nuclear hydrogen requires no fossil fuels, results in lower greenhouse-gas emissions and other pollutants, and lends itself to large-scale production. These advantages do not ensure that nuclear hydrogen will prevail, however, especially given strong competition from other hydrogen sources. There are technical uncertainties in nuclear hydrogen processes, certainly, which need to be addressed through a vigorous research and development effort. The hydrogen storage and distribution are also important area of research to be undertaken for bringing in a successful hydrogen economy regime in future.

The current worldwide hydrogen production is roughly 50 million tonnes per year. Although current use of hydrogen in energy systems is very limited, its future use could become enormous, especially if fuel-cell vehicles would be deployed on a large commercial scale. Meanwhile in the near term, the developments of plug-in vehicles and hybrid vehicles could provide enough experience on the use of hydrogen in the transport sector. The hydrogen economy is getting higher visibility and stronger political support in several parts of the world. In addition, in a future energy economy, hydrogen could compensate for the variable demand for electricity as a storable medium by means of fuel cell power plants and also serve as spinning reserve. Together, they both offer much more flexibility in optimizing energy structures.

Considerable work has been done regarding technologies for the nuclear production of hydrogen, and technical feasibility is well established. Significant issues remain with regard to the development of licensed, economically competitive designs, but the enormous energy market associated with transportation alone justifies the investment of funds required to address these issues to enhance the efficiency of hydrogen production in the long term. In the nearer term, production of hydrogen through electrolysis using nuclear-generated electricity can be a viable option, particularly for the distributed production of hydrogen using off-peak power. The US, Japan, and other nations are exploring ways to produce hydrogen from water by means of electrolytic, thermo-chemical, and hybrid processes. Most of the work has concentrated on high-temperature processes such as high-temperature steam electrolysis (HTE) and the sulphur–iodine (SI) and calcium–bromine cycles. These processes require higher temperatures (>750°C) than those that can be achieved by water-cooled reactors.

Advanced reactors such as the very high-temperature gas-cooled reactor (VHTGR) can generate heat at these temperatures, but will require several years before they are commercially deployed. Yet, high-temperature reactors are seen as the most suitable option for the production of nuclear hydrogen using either the sulphur–iodine thermo-chemical cycle or high-temperature electrolysis. Current light water reactors represent another approach for the production of nuclear hydrogen when their off-peak nuclear-generated electricity is being used with existing water electrolysis production technologies.

9.5.3 Other industrial applications

The main industries interested in the use of process heat are the petroleum and coal processing, chemical, paper, primary metal and food processing industries. The application of nuclear heat for industrial process applications has significant potential that has not yet been realized to a large extent. Currently, only the Goesgen reactor in Switzerland and the RAPS–2 reactor in India continue to provide heat for industrial processes, whereas other nuclear heat systems used for industrial processes have been discontinued even after successful use. Among the reasons cited for the closure of these units, one is availability of cheaper alternate energy sources, including waste heat near the industrial complexes. Previous experiences with nuclear energy in providing process heat for industrial purposes exist in Canada, Germany, Norway and Switzerland. In Canada, several CANDU reactors supplied steam for industries such as food processing and industrial alcohol production until 1998. In Germany, the Stade PWR has supplied steam for a salt refinery located 1.5 km from the plant during the period December 1983 to November 2003. In Norway, the Halden Reactor has supplied steam to a nearby factory for many years. In Switzerland, the Goesgen PWR has been delivering process steam to a cardboard factory located 2 km from the plant since 1979.

9.5.4 District heating

Economic studies generally indicate that district heating costs from nuclear power are in the same range as costs associated with fossil-fueled plants. In the past, the low prices of fossil fuels have stunted the introduction of single-purpose nuclear district heating plants. Although many concepts of small-scale heat-producing nuclear plants have been presented during the years, very few have been built. However, as environmental concerns mount over the use of fossil fuels, nuclear-based district heating systems have potential. As will be shown, there is indeed a very large market for district heating. Nuclear district heating is in use in several countries and is technically a mature industry. District heating accounts for 11% of total final energy consumption in Central Europe and Ukraine and over 30% in Russia and Belarus. District heating accounts for almost half of the heat market in Iceland (95%), Estonia, Poland, Denmark, Finland and Sweden. Its future expansion will be determined by a combination of several factors, such as the size and growth of the demand for space and water heating, competition between heat and non-heat energy carriers for space and water heating, and competition between nuclear and non-nuclear heating. The availability of a heat distribution network is an important factor for nuclear district heating. In technical perspectives, district heating requires a heat distribution network for transporting steam or hot water with a typical temperature range of 80–150°C, a heat source in a range of 20 km close to the customer, a small capacity of 600–1200 MW(th) depending on the size of the customer, an annual load factor of less than 50%, and the required backup capacity.

9.5.5 Innovative applications

Another potential future application of nuclear process heat is the use of nuclear energy for fuel synthesis (including hydrogen production), coal gasification, and oil extraction including oil sand open-pit mining and deep-deposit extraction in Canada. Alberta's oil sand deposits are the second largest oil reserves in the world, and have emerged as the fastest growing, soon to be dominant, source of crude oil in Canada. Coal gasification/liquefaction as a relatively cleaner fossil fuel source are an area of active interest. Production of synfuels and other hydrocarbons using nuclear heat is another area of greater promise. CO_2 can be used as feedstock together with water, nuclear heat and electricity for producing synthetic hydrocarbons, which may be a better energy carrier than hydrogen. This can also act as a CO_2 sink, reducing its emission to the environment. Preliminary estimates indicate that synfuels could be produced at prices comparable to or even lower than those of fossil fuels. Further work on integrated nuclear–chemical complexes is desirable to gain vital experience in this area. Hydrogen may be applied to all types of transportation including aircraft, ships and trains (all could be powered by liquefied hydrogen). Future widespread use of gaseous hydrogen for fuel road vehicles is already widely acknowledged.

9.6 Sources of further information and advice

IAEA Advanced Reactors Information System (ARIS), http://aris.iaea.org

The IAEA has developed ARIS (IAEA, 2010), a web-accessible database that provides Member States with balanced, comprehensive and up-to-date information about all advanced reactor designs and concepts. ARIS includes reactors of all sizes and all reactor lines, from evolutionary water-cooled reactor designs for near-term deployment, to innovative reactor concepts still under development such as gas-cooled and fast-reactor designs or small- and medium-sized reactors. ARIS allows users to sort and filter the information based on a variety of relevant criteria, thus making it easy to capture the general trends and to identify the differences between the diverse designs and concepts.

Generation IV International Forum, http://www.gen-4.org/

The Generation IV International Forum (GIF, 2002) is a cooperative international endeavor organized to carry out the research and development needed to establish the feasibility and performance capabilities of the next-generation nuclear energy systems. Their website provides technical information about the innovative reactor concepts being considered under GIF.

IAEA Nuclear Hydrogen Production, http://www.iaea.org/NuclearPower/HEEP/

In addition to several publications on hydrogen production using nuclear energy, the International Atomic Energy Agency makes available at no cost to all Member States software called the Hydrogen Economic Evaluation Programme (HEEP).

IAEA Nuclear Desalination, http://www.iaea.org/NuclearPower/Desalination/

In addition to several publications on nuclear desalination, the International Atomic Energy Agency makes freely available to all Member States software termed the Desalination Economic Evaluation Programme (DEEP).

9.7 References

EPRI (1995, 1999), *Advanced light water reactor utility requirements document*, Volume I, Rev. 1, December 1995; Vol. II and III, Rev. 8 (March 1999), EPRI, San José, CA.

EUR (2001), *European utility requirements, http://www.europeanutilityrequirements.org/*, EUR, Paris.

GIF (2002), *A technology roadmap for Generation IV nuclear energy systems*, GIF-002-00, US Department of Energy, Washington DC.

IAEA (1997a), *Terms for describing new advanced nuclear power plants*, IAEA-TECDOC-936, IAEA, Vienna.

IAEA (1997b), *Objectives for the development of advanced nuclear plants*, IAEA-TECDOC-682, IAEA, Vienna.

IAEA (1999a), *Introduction of small and medium size reactors in developing countries*, IAEA-TECDOC-999, IAEA, Vienna.

IAEA (1999b), *Evolutionary water cooled reactors: Strategic issues, technologies and economic viability, Proceedings of a symposium held in Seoul, 30 November – 4 December 1998*, IAEA-TECDOC-1117, IAEA, Vienna.

IAEA (2002), *Improving economics and safety of water cooled reactors: Proven means and new approaches*, IAEA-TECDOC-1290, IAEA, Vienna.

IAEA (2010), *Advanced Reactors Information System (ARIS)*, http://aris.iaea.org, IAEA, Vienna.

10
Nuclear safety in nuclear power programs

D. A. MENELEY, Atomic Energy of Canada Ltd, Canada

Abstract: Nuclear safety includes all aspects of protection of humans and the environment from the harmful effects of ionizing radiation existing or produced during operation. This chapter outlines all aspects of the safety of nuclear-electric generating stations with the exception of conventional industrial safety. Due to the very broad scope of this subject, extensive reference is made to open literature on the subject. The objective of the chapter is to assist those persons interested in starting a new energy venture to reach a basic understanding of this technology and its application to satisfy human needs for energy.

Key words: protection of the public, international standards and guides, national regulatory body, operational safety, safety management systems.

10.1 Introduction

Close attention to nuclear safety is easily justified on each of three factors: protection of the public, protection of the operating staff, and protection of the plant. As identified in governing regulations (IAEA, 2006), safety is the full responsibility of the plant licensee (INSAG, 1999a, p. 15). From first principles, any delegated responsibility still remains in full force with the delegator. Each regulatory agency acts in the role of safety auditor during operation in order to ensure that the plant is operated within the scope of the licensee's authority and in accordance with national standards and regulations. The operating organization holds, at all times, authority to operate the plant only within the provisions of the operating licence, and commensurate with its stated responsibility. This authority normally is delegated by the regulatory agency on behalf of the government of the country. Since delegated responsibility *always* remains in full force with the delegator, the government and regulatory agency remain ultimately responsible for safe performance of the nuclear energy enterprise.

10.1.1 Protection of the public

This factor usually receives the most attention in day-to-day discussions because of its importance within the political process and therefore the emphasis on public protection by the safety regulatory agency. The owner must, of course, *justify* adequate safety to the regulatory authority in order to obtain permission to operate the plant. It also is necessary for the owner to operate the plant safely at all times, in order to maintain the trust and goodwill of the community. This illustrates a fundamental reality; it is that any regulatory agency in a nation with a responsive government must conduct a licensing process that is partly technical and partly socio-political – and the ultimate judge of sufficient safety is the body politic.

10.1.2 Protection of the operating staff

Normally, the plant owner must *justify* sufficient nuclear safety provisions to satisfy the requirements of labour regulatory bodies as well as those deemed essential by union and non-union staff. This requirement becomes an integral part of the regulations related to safety in the workplace, also known as industrial safety. The owner must train and maintain the vigilance of staff exposed to ionizing radiation in the course of their duties.

10.1.3 Protection of the plant

Protection of the plant normally is not considered in discussion of safety principles. However, all safety authorities recognize the importance of a healthy safety culture to maintaining low plant risk (i.e. excellent plant safety). All safety culture begins with senior management. Protection of the plant investment is widely recognized as one of the fundamental responsibilities of senior management, usually by the plant's shareholder(s). The operating organization must *justify* plant protection to the owner(s). Over time, congruence of these two management responsibilities may prove to be the single most important factor in assurance of real safety within and outside the plant at all times.

Protection of the plant includes protection against damage from external hazards. In the first instance, this falls within the scope of the owner's investment protection – for example against fire, flood, wind, earthquake and other natural phenomena. Protection of plant *functions under these conditions* is, of course, to be considered as one aspect of public and staff protection.

10.1.4 Scope of application

Consideration of nuclear safety begins on or before the first day of a proposed project; this chapter is devoted entirely to a description of this aspect of plant justification. At the same time it must be recognized that all of the earlier considerations lead up to safety in actual operation – the plant is safe in the sense of radiological risk until its nuclear fuel materials arrive on site. From that day until the day that the licence of the plant is finally transferred to a second licensee or until the need for a radiation-related-licence is no longer required, it is the operating organization that is ultimately responsible for its safety, within the scope of its operating licence and in accordance with national standards and regulations.

10.1.5 Support systems for the operating company

There are many organizations that can and will help an operating organization build up the necessary skills, recommend their optimal business and professional infrastructure, and assess the performance of the organization over the lifetime of the plant. In addition, current members of some of these organizations, usually who operate similar plants to the one that the new operating organization has built, have formed 'owners groups' with the purpose of exchanging detailed operating information and experience. Some have set up arrangements whereby they manage common research and development projects on behalf of their members. These owners' groups have proven to be very valuable in broadening knowledge as well as in reducing operating costs.

10.1.6 Independent safety auditor

From the earliest beginning of the nuclear era, governments have established, and then have relied on regulatory organizations to audit the performance of organizations of all sorts related to use of ionizing radiation – isotope users, miners, researchers, health professionals, and power plant operators. These regulatory organizations issue licences to operate within carefully defined rules and regulations. They usually perform detailed auditing and enforcement duties, especially through staff members at the location of major facilities such as power plants.

The positive value of audit staff to the plant owner/operator arises from their emphasis on safety. This emphasis helps to provide balance to the strong motivation of plant senior management, who may at times consider production as their first and overriding priority. This need for balance provides the most fundamental infrastructure requirement that justifies the purchase of a nuclear plant; that is, the need for a competent review of plant safety performance *before* the plant is purchased, to ensure that later performance will meet the exacting standards required by the safety regulatory agency.

10.2 Basic safety principles

The basic safety principles applicable to nuclear power plants have been well known for decades. They were first documented by the IAEA

International Nuclear Safety Advisory Group (INSAG, 1988) and were updated in 1999. In 2006, the IAEA published a broadly supported document entitled *Fundamental Safety Principles* (IAEA, 2006). These documents stop short of defining the requirements of any specific power plant design, but do provide a prospective plant owner with all of the principles on which the modern world's nuclear safety approaches are based.

10.2.1 Protection required for large and small nuclear reactors

The most important information regarding public safety is to determine both the frequency and the consequence of any potential reactor accident. The consequence of a severe reactor accident could, for example, be an excessive dose of radiation to one or more members of the public. The risk of such an event depends, of course, on the amount of radioactive material that could be released – that depends, other factors being equal (e.g. reactor and containment design), on the power level of the reactor under consideration. Smaller reactors do not have the same level of radioactive material in them, and so their inherent risk is lower.

Large power reactors

At the beginning of the nuclear energy era, several countries developed more or less independent standards in the attempt to define the acceptable level of risk, in terms of risk to the general public. The methods varied from simple dose limits within a specified frequency range (Hurst and Boyd, 1972) down to detailed listings of specific equipment that must be installed (Murley *et al.*, 1991). These criteria have been refined over past decades; the present-day international standards for achieving a satisfactory level of power reactor safety can be found in IAEA publications (NS-R-1, 2000 and 13 associated guides). These documents are not, in general, specific enough to serve as national standards for reactor licensing. Generally, each national regulatory group establishes a unique set of specific documents for licensing purposes. Most of these national regulations make reference to the higher-level IAEA documents. Within the community of large reactor owners there has been a major cooperative effort to establish inter-plant communication as well as codes of 'best practice' to disseminate detailed plant operational and safety information to new owners. It is strongly recommended that a prospective new plant owner should join the appropriate group and to seek information and training from them. This in recognition of the large financial benefits that can be gained by doing this, as well as the fact that nuclear plant owners know that a serious accident anywhere in the world has immediate deleterious effects on all operating plants.

Small power reactors and research reactors

Recently, IAEA has produced a separate Safety Requirements document (NS-R-4 and six associated guides) for small reactors. The Agency is preparing one further guide entitled *The Graded Approach*. This guide will describe the unique acceptance criteria for this category of fission reactors. In the meantime, at least one national regulatory agency (Canada) has already produced such a guide showing details of the different levels of requirements that may be applied in licensing of small reactors, as well as giving a definition of this reactor classification.

10.2.2 Safety management

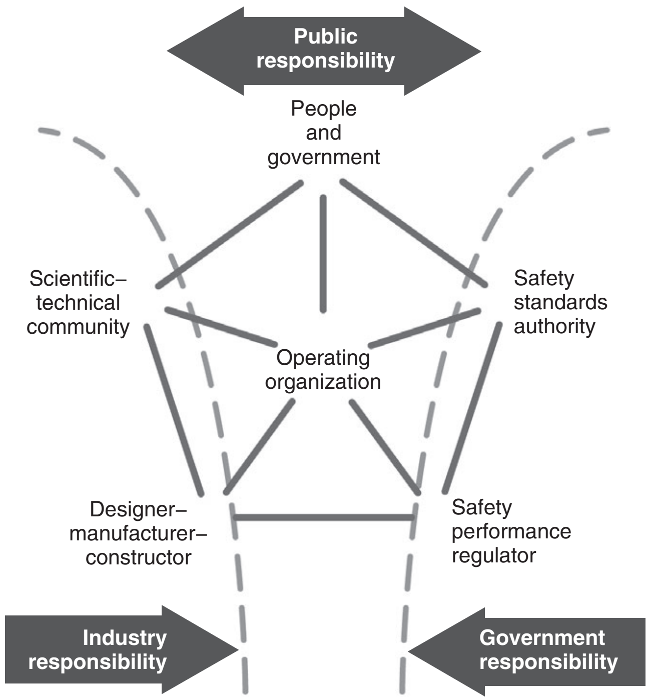
Senior management of any organization whose operation has been authorized by a national regulatory body must be knowledgeable of its obligations under the operating licence and must be directly involved in setting and sustaining safety policy of the organization (IAEA, 2006). It has been shown that effective leadership is required to sustain a high level of performance, because senior management sets the 'tone' for the whole organization.

Early in the process of starting a nuclear program, governmental authorities of the country must establish a stable relationship between several aspects of the program. Figure 10.1 presents one model of a stable safety management relationship.

Safety culture – human performance

The concept of safety culture was introduced to the nuclear industry in 1986, shortly after the disastrous accident at the Chernobyl Unit 4 in the Ukraine. The concept originated in the chemical industry, where it had been shown to enhance human performance by emphasizing the individual responsibilities of operators for the safe performance of the facility.

The International Nuclear Safety Advisory Group (INSAG) has issued two documents, INSAG-4 (1991) and INSAG-15 (2002), on the subject of safety culture. Safety culture is included as one of the important management principles (IAEA, 2006, p. 8). Safety culture has been defined many times in several different ways. On the other hand, the idea itself seems to be quite simple – it is a methodology intended to maximize human performance. Figure 10.2 represents a cycle of human performance that we all can recognize. Some days we feel bright and confident, ready to deal with any situation that arises during our working day. On other days we feel that we are (almost) superhuman. Then, there are days when boredom sets in, and nothing seems to be worth doing. And lastly, there are days when we are unsure of our ability to perform any complex task in a competent fashion.



10.1 The Safety management system.

Now, imagine a group of people trying to complete a cooperative task. Some happen to be at the high point in their cycle and others at the low point. The supervisor's job description says that he is to make the whole group perform in the upper-right quadrant of the cycle; that is, with safety and confidence. However, he or she is also human, and subject to the same cyclic behaviour.

If the group is 'in sync', and all are operating in the upper right quadrant, a great deal of valuable work can be done. But if the group is 'in sync' and all are at a low point in their personal cycles, then the whole group is ineffective, and possibly unsafe.

The job descriptions often referred to as 'senior management' can exert a powerful influence on this success–failure cycle. They can consistently encourage staff to work up to their best potential and thereby tend to keep them in the high-performance category, or they can discourage staff by their own attitude, job performance, or opinions of their work groups they periodically express.

'Safety culture' refers, therefore, to a complex matter involving human behaviour in groups. Since human beings are by far the most complex element in any power plant, it is vital to study and maintain sound

10.2 The human cycle of performance.

methodologies and organizational infrastructure that work best within the larger social culture of the local community. This subject is at least as important to safe operations as are the hardware and equipment installed in the plant.

10.2.3 Defence in depth and defence in time

The concept of defence in depth was first utilized in military practice. Scattered defensive outposts are, of course, vulnerable to defeat through local attack. To overcome this weakness, the defence in depth idea established methodologies through which the outposts were linked together via communication channels and response doctrine that specified assistance from one outpost to others nearby, along with an established deep configuration of outposts that together formed a strong network of defence.

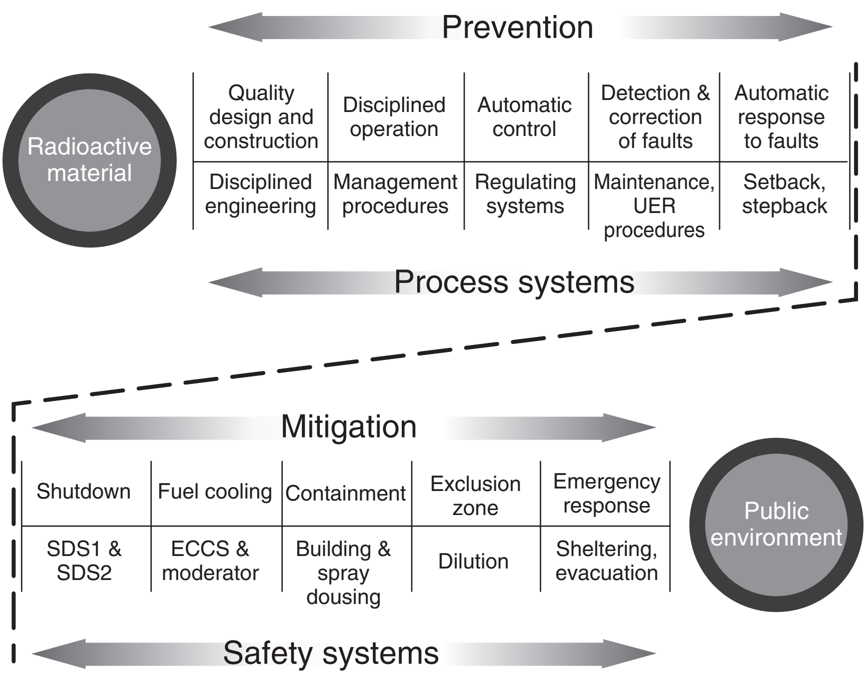
Defence in depth

This useful concept has been adopted by the world nuclear industry. A good summary of the application of this concept can be found in the IAEA report titled *Defence in Depth in Nuclear Safety* (INSAG-10, 1996). This document is based on the original description of this concept published in an earlier document (INSAG-3, 1988). Similar descriptions have appeared in design and safety-related documents published over the past few decades. Figure 10.3 illustrates the overall concepts of defence in depth. In this view, the processes are separated into two parts – prevention and mitigation, respectively. Some reactor designs may have different specific elements in some of these positions; however, the principle remains the same – there are multiple levels of defence against transfer of radioactive materials from their normal positions in the reactor to the public or environment.

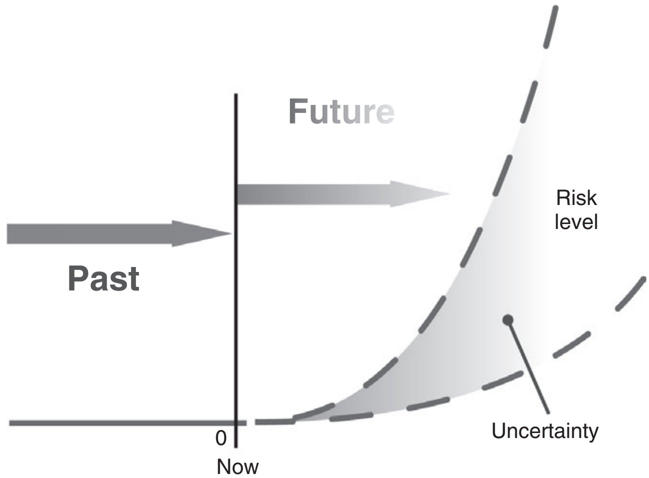
Defence in time

The concept of defence in time is much less widely accepted. However, the components of defence in time are included in many publications related to operational safety. An excellent description of operational safety principles and practice is presented in the report *Management of Operational Safety in Nuclear Power Plants*, INSAG-13 (1999b).

Figure 10.4 illustrates the need for defence in time. The question of needed defence begins in the immediate present. We can presume that, at this time, all plant systems are performing perfectly, in accordance with the requirements of the operating licence and in accordance with the design



10.3 The defence in depth concept.



10.4 The need for defence in time.

intent. Now, as the time interval beyond this instant increases, uncertainties will arise with respect to the functionality of components and systems. The future is inherently uncertain. The direct question may be 'Should we do inspection or maintenance operations of component or system "X" at this time, or can it wait until tomorrow?' As time passes the overall uncertainty increases regarding the plant's performance under both normal and potential abnormal operating conditions – sometimes very rapidly. The answer, of course, is careful monitoring of all systems, inspection, and maintenance. These multi-faceted actions together constitute 'defence in time'.

Obviously, the operating crew must carry the responsibility and authority for this aspect of safety defence. Infrastructure and methodologies for carrying out these tasks must be established before plant first begins to operate, and must be continued for the whole lifetime of the plant.

An integral part of defence in time is regular examination, throughout the life of the plant, of events in the environment around the plant and to some extent events in the whole world that might reveal important shortcomings or unappreciated advantages of the plant for which the operating crew is responsible. Revisions and upgrades may be initiated based on these regular examinations.

10.2.4 Safety responsibilities and authorities

A mature nuclear energy system includes a large number of people and related organizations. However, one finds that there is a common basic structure needed for successful conduct of any program. This is here identified as the safety management system.

Safety management system

Maintaining and improving nuclear energy's safety record requires careful attention to authorities and responsibilities so that safety responsibility is always placed only along with commensurate authority. Proper assignment of responsibility and authority is the very foundation of safety. Note that a very similar description of the roles of various groups in plant operation can be found in INSAG-13 (1999b).

The fact that safety is an operational matter places the operating company in the central position of safety responsibility. Figure 10.1 shows the relationship between the operating company and the other major participants. This is *not* an organization chart, but is used to indicate the relationship of authority and responsibility between the main participants. The top reporting relationship is to the public. Supporting roles are played by technocrats on one side and by bureaucracies on the other side – the safety tribunal authority and the safety performance regulator.

Role of the designer/manufacturer/constructor

The newness of nuclear generation technology, the dominant place of the designer/builder in development of the system, and the extended process of plant design and construction before the first operating licence is issued tend to leave the impression that the central role is played by this group. However, it must be recognized that the designer/builder leaves the site shortly after first operation and has (at least in Canadian practice) no further responsibility for the plant, following handover from the vendor to an operating organization. Similar handover practices exist in most, if not all, countries. For example, in France where EDF is heavily involved in the building of nuclear plants as well as serving as the sole operating organization, handover of the plant must be formally executed from the building unit of the company to the operating unit of the same company. The principle remains the same. The primary role of the designer/builder is to deliver a plant to the operating organization that not only meets regulatory requirements but also meets the staff and plant protection safety goals. During the operating phase, particularly in the early years, the designer/builder might perform support services to the operating organization. These services must eventually be taken over by either the operating company or a related organization whose only commitment is to support of operating stations. The operating company has a return responsibility to the designer/builder. It must inform designer/builder staff of the design features that are most useful during operation from the point of view of performance and safety, in addition to comprehensive operational feedback on any components or systems that require improvement. Practical considerations will vary in each individual case; in every case the essential linkages that must be sustained over the whole operating life of the plant are illustrated in Fig 10.1. Fortunately, in recent years it has become 'best practice' for the designer/builder to transfer to the operating organization the same comprehensive CADD (computer-aided drafting and design) model that was used to construct the plant. Associated materials lists and other supporting documentation is also transferred to the operating organization, to serve as a complete record of the facility 'as built'. This model then can be used by the operating organization to maintain a record of all in-service experience and maintenance operations. One alternative method for retention of these data is described in another INSAG report (INSAG, 2003).

Safety performance regulator's role as auditor (regulatory staff)

The regulatory staff is assigned the auditor's role by the safety standards authority. They review design features, operating procedures, and training to determine the acceptability of the plant for initial and continued operation. They have no role in design or operation. Furthermore, they cannot take any such role without compromising their position as impartial auditor. The auditor's role involves a great deal of questioning of the operating company and designer/builder on details of design and operation. This role is never a popular one, particularly when approval to proceed with some action is held up, apparently to satisfy curiosity. There is, no doubt, some unnecessary holdup caused by lack of understanding or by personal factors. One the whole, the process is useful to the operating company because this is the only external and independent (not to say hostile) review of proposals. Internal reviews are valuable but sometimes miss important issues due also to lack of understanding or to personal factors.

One of the most valuable early decisions of the Canadian AECB was to assign staff at each station site. These people get to know a particular plant as well as the operating company supervisory staff, and often much better than the designer/builder or central office staff. They are therefore able to make reasoned judgments of the quality of safety-related aspects of plant operation on a regular basis. Knowing both the equipment and the people, they are better able than are central office staff to evaluate special situations that arise in the field. Central office staff are useful as technical backup, but the site staff must carry the main regulatory responsibility. The operating company has an obligation to report matters of safety interest to the regulatory staff on a regular basis as well as to report any unusual occurrences.

Role of the safety standards authority (the tribunal)

The safety standards authority – in the Canadian case the Nuclear Safety Commission – carries the authority delegated from the government (and ultimately from the people) to administer the Nuclear Safety and Control Act (CNSC, 2000). This Act grants very broad powers to make regulations for the administration of the Act. Up until recently, the CNSC chose to write only general regulations; specific regulatory requirements were applied through the licensing process – and so were largely determined by the regulatory staff.

In general, the role of the tribunal is to determine the rules under which radioactive materials and processes must be managed in Canada. With regard to any activity involving ionizing radiation, they sanction the game; that is, they permit the activity to proceed provided that the rules are followed. Their ultimate power is to stop the activity if the rules are violated.

Role of the scientific/technical community

The group is defined in terms of professional standing. The operating company may employ some members of this group, while others may report to organizations such as governments, engineering companies, research laboratories, and universities. Their common goal is to establish and maintain the scientific and technical information necessary to carry on the nuclear enterprises. In addition, it is their responsibility to carry on their activities within the bounds of high professional and ethical standards. On occasion, these goals come into conflict with some of the goals of the organizations in which these professionals are employed, particularly in matters of judgment on the importance of particular technical facts. In such cases their employer must recognize the requirements of professional conduct under which the scientific/technical group operates.

The scientific/technical group assists the designer/builder and the operating company in defining the equipment and procedures necessary to achieve safe operation. This group also deals directly with the public in explaining the details of nuclear power technology and answering any concerns that they express. In our society, the scientific/technical group has a very high rating of credibility with the public. This trust rests, of course, on their continued adherence to the high professional and ethical standards noted above. One this credibility is lost it can be very difficult to recover. This is one reason that employers must recognize their need to speak openly and honestly in areas of their own professional competence. The scientific/technical group must also recognize their special position as trusted interpreters of technology to the public. In recent years there have been many cases in which members of this group misused this trust by making unsubstantiated claims on one side or the other of the nuclear power controversy. The overall effect has been a reduction in the credibility of this group with the public. In summary, the major roles of the scientific/technical group are (a) to provide reliable technical data for design, operation, and licensing, and (b) to inform the public of the realities of nuclear energy technology.

Role of the public and government

In the Canadian political system, as well as in many other nations, the public ultimately decides what is to be done and what is to be stopped. In this sense the whole of the nuclear enterprise reports to them. Officially, this reporting is done through government agencies and elected officials. In recent years, however, the public has become much more directly involved – the system has become more participatory and less representative. We all can recall cases in which public discussion has directly influenced the decisions made by both the operating company and the safety standards authority. The safety management system has become a political system rather than a purely technical one. This subject is discussed in some detail in a recent report by INSAG (INSAG, 2006).

In the present-day climate, consider the position of the operating company when faced with a regulatory staff proposal with which they disagree, either on the basis of potential negative effect on safety or due to unfavourable cost-effectiveness. They can appeal this proposal to the safety standards authority in hopes that reason will prevail. The safety standards authority may rule against the operating company at least partly because of their heavy reliance on the regulatory staff for technical advice. Several means have been devised to ensure that regulatory decisions (which may have far-reaching consequences) are balanced. The first is to establish a senior advisory committee reporting to the head of the regulatory agency, whose duty is to advise the authority from a detached, third-party point of view. In some countries, formal appeals can be made to separate and unbiased bodies established for this purpose.

Role of the operating company

Referring back to Fig. 10.1, the operating company may be considered either to be at the centre of the action or to be surrounded on all sides. The one indisputable fact is that the operating company is in the nuclear energy business for the long run. Since the station is already committed and running, the capital is spent and a favourable return on investment can be obtained only by operating the plant for a number of years, the operating company has no way out but straight ahead. The key element for success of the enterprise is for the operating company to earn the confidence of the people in their ability to run the plant safely and efficiently.

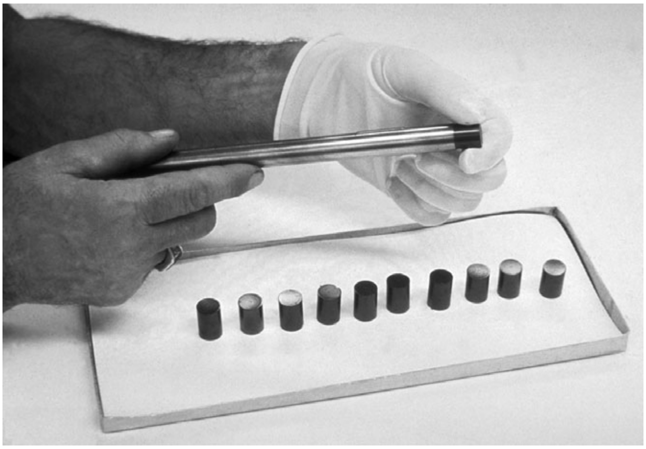
10.3 Development and application of deterministic safety assessment

First, we must ask the question: 'Why are nuclear reactors hazardous, and in exactly what way are they hazardous?' The answer to this question (Meneley, 1999) should underlie the rationale for all analysis of potential failures and the means for mitigating those hazards.

10.3.1 Hazards of solid fuel reactors

A typical fuel pellet made of sintered uranium dioxide (melting point approximately 2800°C) is shown in Fig. 10.5. Millions of such pellets are located in an operating power reactor. Almost all the reaction products of fission – fission products – are trapped inside this pellet. The second important fact is that most of the heat energy of the fission process is produced inside this same pellet. Heat is removed by flowing coolant (usually high-pressure water).

It is apparent that either increasing the rate of heat production (i.e. increasing the rate of fission) or decreasing the rate of heat removal (i.e. decreasing local water pressure or flow) threatens to increase the temperature of the fuel pellet and therefore bring it closer to melting temperature. If and when the pellet melts it will release essentially all of its fission products that are volatile at the mixture temperature – some of which are highly radioactive. These fission products represent the main hazard of nuclear reactor operation.



10.5 Typical nuclear fuel pellets.

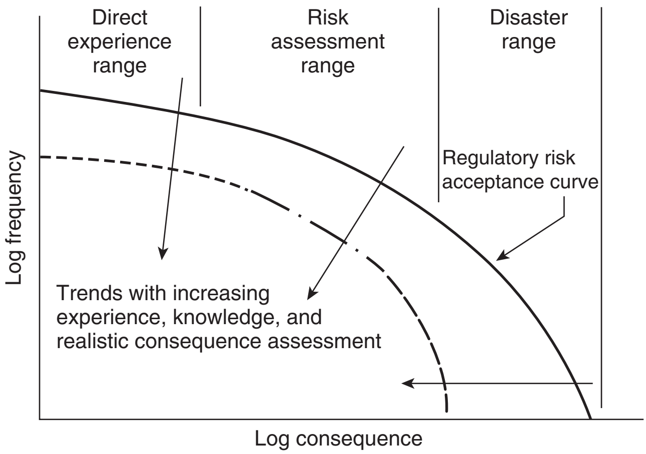
High-pressure water presents an obvious hazard due to the possibility that a pipe might break and release the water, and so might lead to overheating of the fuel pellets, if emergency water supply were not available. Other initiating events that reduce the heat removal rate (e.g. loss of forced circulation) add similar hazards.

Returning to the fission process itself, and recalling that the process involves a chain reaction, sheds light on yet another hazard of fission reactors. We all know that a chain reaction involving successive generation of fissions is at the heart of this technology. We also know that a few (less than 1% of the total) of the next-generation neutrons essential to keep the chain reaction going at a constant rate are emitted after a slight delay. This is known as the 'delayed neutron fraction'. Further, to increase reactor power we must manipulate controls so that the number of neutrons in each successive generation is slightly larger than the number in the previous generation. It is important to control this increasing neutron population to a very low rate so that engineered control systems can return the excess number of neutrons per generation to zero once again, when the desired neutron density (proportional to the reactor power level) is reached.

A serious hazard may arise if and when the excess number of neutrons in successive generations approaches or even exceeds the number of delayed neutrons in that generation; in such a case the rate of multiplication becomes very much faster. If this number exceeds the delayed neutron fraction, the dominant rate of power increase becomes inversely proportional to the time between successive fissions, and the delayed neutrons are left behind. This is known as the 'prompt critical' state. Different reactors have different characteristic times; they range between about 1 millisecond in a thermal neutron reactor design to less than a microsecond in a fast neutron reactor. In every case of abnormal operation when a 'prompt critical' state can occur it is vital to ensure that either inherent characteristics or highly reliable engineered systems will act to return the reactor to a non-self-sustaining condition – that condition is known as the 'shut down' state.

10.3.2 Development of safety protective logic

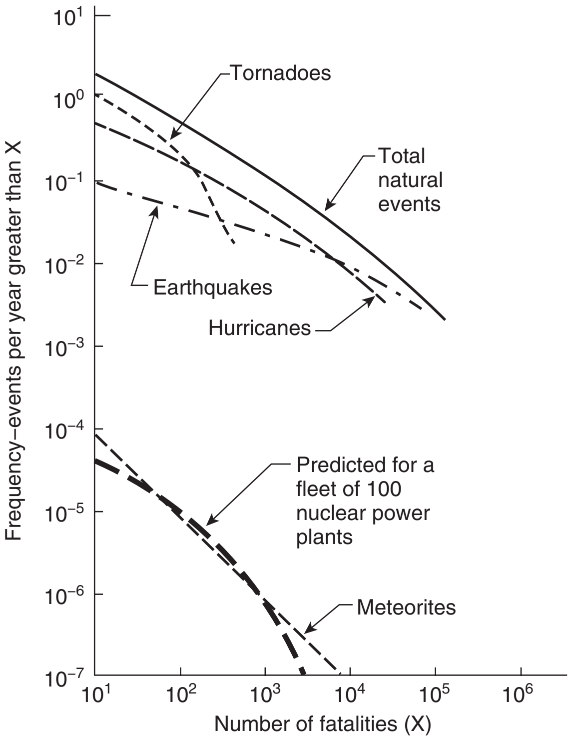
In early years, two quite distinct approaches to safety design and licensing were developed. The first has been associated with Frank Farmer (Farmer, 1967), who argued that the fundamental rule of engineering design requires recognizing the desirable inverse relationship between accident frequency and expected accident consequences. This method was elaborated by E. Siddall and others and then applied to the licensing of the first large-scale CANDU power plant. In this formulation, accidents of all types can be presented (Meneley, 1999) on a frequency versus consequence plot (Fig. 10.6). The initial Canadian approach was later modified to an intermediate



10.6 Risk curves and trends.

method combining the initial probabilistic formulation with specific requirements to be applied separately to systems used to operate the plant, and secondly to an independent set of so-called Special Safety Systems whose only functions were to respond to abnormal conditions so as to shut off the chain reaction, close the containment envelope, and continue cooling the fuel. The current licensing regime in Canada continues in this same style, even though many detailed requirements have been added to the original concept.

The second approach to licensing was to first establish a set of so-called General Design Criteria for Nuclear Power Plants (USNRC, 2010) and then to judge licence applications in terms of their success in meeting these criteria. In this approach there was no explicit appeal to accident frequency, even though the underlying logic can be interpreted as such. This approach is still used in the USA and in many other countries; however, it has been augmented in many respects, especially through the introduction of specifications requiring detailed probabilistic analytical tools. This probabilistic approach builds on the work presented in the original report (Rasmussen *et al.*, 1975). Figure 10.7 shows a very brief indication of the original results. Note that it estimated the risk of operation of 100 large nuclear plants to be similar in magnitude to the existing risk of fatalities caused by meteorite strikes. Other naturally occurring risks were found to be several decades larger. In spite of this highly reassuring finding, fear of nuclear energy has for more than 25 years barred the further adoption of this safe, economical, and sustainable energy source in the United States, and largely in Western Europe. There is a fundamental lesson in this experience for nations that choose nuclear energy and seek to justify that choice to their citizens (see Section 10.3.4).



10.7 Frequency of natural events involving fatalities (USNRC, 1975).

The extensive background experience in setting and then meeting safety standards in operation, as described in this section, has produced a well-codified set of international standards and guides for safe operation that can be used with confidence by organizations ready to join in the worldwide nuclear energy enterprise.

10.3.3 International standards and design guides

Over the past several decades the IAEA has utilized the technical expertise of its member states to formulate and publish standards related to all aspects of nuclear reactor safety. These standards can be used immediately by the regulatory authorities of member states, to establish their own unique safety regulations as befit their unique circumstances and to incorporate these regulations, as desired, into their national governance structure.

The standards and guides developed by the IAEA have been augmented over the years by the work and publications of the International Nuclear Safety Advisory Group, appointed by the Agency's Director General.

Because of their concise format and their high standard of intellectual integrity, this summary of the international safety regime uses these INSAG documents as the primary source of guidance to new plant users.

Severe accidents

One of the most important conclusions in the INSAG-12 report (INSAG, 1999a) is the upper frequency target of 10^{-4} or less per year for severe core damage. This principle is discussed under the heading 2.3 Technical Safety Objective, paragraphs 19 to 27, in the INSAG-12 document. The objective of this approach is to establish an upper bound recurrence frequency of severe core damage accidents that could lead to release of a large amount of radioactive material that could, if the containment structure were damaged consequent to the event, lead to severe consequences for human health.

External events

This term is used to identify abnormal events that are initiated from outside the nuclear station. Some examples are earthquakes, tornadoes, aircraft crashes, and floods. The plant can be protected from many of these events before the fact by careful location and investigation of the potential consequences of their occurrence combined with the estimated probabilities. The designer is expected to provide either passive or active defence against such events; the acceptability of these provisions is one of the major components of regulatory review for approval of a nuclear station site.

Recent events (e.g. earthquakes and tsunamis at the plants at Kashiwazaki and Daiichi in Japan) have underlined the importance of seismicity in planning for location of a nuclear station, and for the facilities needed for its protection. Specifically, the Daiichi situation highlights the hazards of the operating state known as 'station blackout', meaning the total loss of electrical power for an extended time period. Other external events may prove to be equally important in different situations.

10.3.4 Public acceptance

The first and most important issue in establishing a new nuclear power program is *trust*; trust that is earned, deserved, and maintained by all people in the industry. The single, most corrosive factor in relationships between the industry and the public is a lack of trust. If all stakeholders of this enterprise within any nation can establish and retain a high level of trustworthiness, public acceptance will not be a problem. Energy supply is a very public business, and consequence is very easily linked to the root cause of negligence. A successful nuclear operating organization promotes truthful and open interaction with its staff and with the public – nothing is hidden.

10.3.5 Risk involved in nuclear power plant operation

Examining Figs 10.6 and 10.7, it is apparent that the most frequent abnormal events are likely to have zero health consequences to the public. However, these events can be very expensive to the operating organization, because they often result in power decreases or reactor shutdown. (Most of the electricity production cost arises from debt retirement of the plant capital cost; operating costs are a relatively small part of the total.) Plant management has a very strong motivation toward reduction of these minor malfunctions. Obviously, this reality exerts a strong positive effect on overall plant safety motivation. The mid-frequency range of malfunctions also exerts a strong positive influence on plant safety – there is a possibility that such accidents might result in some damage to the plant, or at least may lead to extended outage time for inspection, repairs or plant modifications. This is the frequency range in which probabilistic safety analysis is most effective, as we will see in Section 10.4.

The 'disaster range' of accident events shown in Fig. 10.6 is the range that concerns safety regulatory agencies the most; it is the range in which the reactor accident might lead to human fatality. In common practice, this is the frequency range (~10^{-4} per year and lower) over which the special safety systems provide the primary defence against health consequences. In US practice, this range is identified as the 'severe accident' range, which includes at least some degree of reactor core disruption.

In the early history of the nuclear industry, this disaster range received an undue amount of attention on the part of designers and regulators. The reason was that accident models were very simple and extremely conservative, so that the predicted consequences were correspondingly large. During the past several years, more exact and realistic predictions of consequences have been made, and so the predicted consequences have become much smaller. Nevertheless, regulatory agencies have tended to continue application of the very rigid and conservative acceptance criteria that were developed during the time when analysis models were crude and overly conservative. The nuclear reactor safety field is now in transition toward more realistic modeling. It is expected that these plants will eventually be proven to be considerably safer in terms of human health than they were originally thought to be.

In the end, then, the question of whether or not nuclear energy will be installed on the very large scale needed to replace today's energy supply from oil, natural gas and coal will probably be determined by a balance of fear – between fear of the technology and the fear of falling short of the high level of supply required to maintain the people of the world in good health and spirit.

10.3.6 Defence in depth, defence in time

Defence in depth is a design philosophy that is applied universally in nuclear reactor design practice. Specific applications differ, of course, as reactor designs are quite different in their needs for protection against various hypothetical events such as sudden closure of turbine shutoff valves, pipe breaks, and accidental control rod ejection. In Canadian design philosophy, for example, each unit incorporates two independent, fully capable and physically diverse shutdown systems to reduce power quickly whenever necessary. There is a fast-acting emergency cooling system that would refill the heat transport circuit in the event of a loss of primary coolant. In addition, the cool moderator water surrounding each fuel channel would remove the decay heat of fission remaining in the fuel, and so prevent fuel melting – as a result, broad dispersion of fission products would not occur. The containment structure features two independent means of sealing the ventilation systems, on receipt of one or more signals. These mechanisms are all kept in a 'poised' condition and are initiated by highly reliable detection and actuation chains with redundant components and 'fail-safe' design characteristics. An exclusion zone surrounds the plant. In this zone no permanent residence is allowed, so that if radioactive materials were to be released in an accident situation there would be no measurable health damage to humans.

Defence in time is a new preventative concept, intended to specifically identify the need for regular attention to the possibility of sudden or wear-out failures of components and systems in use in an operating nuclear station. The basic idea is to establish a methodology requiring preventive maintenance for each component and system important to safety, at time intervals depending on the life expectancy of the item. Regular maintenance ensures that these components and systems are ready to perform their function if required during any possible accident. Testing of these systems is conducted on a regular basis; as a result the system is maintained in an essentially 'as new' condition for the whole operating life of the plant. Other jurisdictions have established similar formal structures, usually via some form of regulatory requirement; for an example, see USNRC (1991).

10.3.7 Adaptability of safety standards to other nuclear technologies

The highest level of IAEA safety standards is intended to be universally applicable to all types of nuclear reactors. At this level the fundamental principles of complex engineered systems and the equally complex field of human interactions are presented; these principles may indeed be applicable well beyond the nuclear power industry.

As the lower level standards reach into matters of greater detail, it becomes necessary to make some of them specific to, for example, water reactors as distinct from gas-cooled or fast neutron reactors. For this reason, and also because the IAEA has no jurisdictional authority over safety regulations within its member states, the international standards stop short of stating requirements for safety of any particular technology – these are left to national regulatory authorities.

In many past situations, the first power reactor of a given type to be installed in a country is purchased from an experienced vendor in a country with an established nuclear program. Normally, the purchase and sale arrangement includes a specification that the plant must be licensable in the country of origin. The purpose of this type of arrangement (sometimes called the 'reference plant' approach) is to foster the transfer of detailed knowledge of the specific technology between an established nuclear industry in the vendor country to the responsible organizations in the buyer's country. This sort of arrangement provides an opportunity for the purchasing country to organize a viable regulatory authority, often with the assistance of both the IAEA and the regulatory staff of the vendor country. It forms a part of a larger contractual issue, usually called the 'technology transfer agreement' between the purchaser and the vendor.

10.3.8 Periodic design reviews and operational reviews

Operational reviews by independent staff (for example, WANO reviews) are conducted to provide station management with an evaluation conducted by independent and experienced professionals. In addition, both national regulatory agencies and the IAEA are available to conduct reviews to ensure that correct operating procedures, training, and maintenance procedures are being followed.

Over the extended operating life of any given plant, new facts may come to light that were unknown at the time of first station operation. When these reviews reveal that some new knowledge has come to light that challenges the overall safety basis of the plant, it may be necessary to install corrective measures or equipment to establish adequate defences.

10.4 Development and application of probabilistic safety assessment (PSA)

As already described, in some reactor safety and licensing regimes the probabilistic nature of this problem was recognized from the very beginning. In those jurisdictions the method developed naturally in parallel with the deterministic method – there was no need for a separate category. For example, the original formulation of the Canadian Siting Guide (Hurst and Boyd, 1972) was augmented with a series of so-called 'safety design guides' (Snell, 2001) that included limited scope probabilistic analysis of each safety-related system (shutdown, containment, fuel cooling) to establish a proof that each of these systems and their support systems (e.g. instrumentation, power, heat removal) could meet the reliability requirement already specified for the particular safety function. In the course of time these PSA components were combined into a single plant-wide safety assessment.

The original regulatory system in the US was closely associated with the so-called 'design basis accident', defined as the set of conditions, needs, and requirements taken into account in designing a facility or product. In nuclear plant design this approach led to simplistic concepts such as 'maximum credible accident', and 'single failure'. Little consideration was given to the many possible sequences of minor events that might combine to result in a major consequence – for example, it took many years before the importance of small breaks in piping was recognized. This early phase of US regulation was superseded by the publication in 1974 of WASH-1400, the Reactor Safety Study, known as the Rasmussen report (Rasmussen *et al.*, 1975). The older requirements for licensing were retained, but much more attention since then has been given to the full scope of potential abnormal events. From this time on, a full probabilistic analysis became an integral part of reactor licensing applications in the US. The associated analytical methods were adopted broadly within the international community.

10.4.1 PSA methods, structures, and limitations

The INSAG report *Probabilistic Safety Assessment* (75-INSAG-6, 1992) presents a sound description of the probabilistic method. It is a systematic risk-based analytical method that combines fault trees and event sequence diagrams of potential success and successive failure pathways, and finally a consequence analysis of each pathway which, when multiplied by the derived frequency of occurrence of the event sequence, delivers an estimated risk (frequency × consequence) contributed by each particular accident sequence. Two additional reports, INSAG-8 (1995a) and INSAG-9 (1995b), provide further details and expansion of the concepts important to probabilistic analysis. The present INSAG group has now published an important extension of this concept as report INSAG-25 (INSAG, 2010).

A PSA analysis may be used to improve a system design by modification to improve the success branch of one or more fault trees. For example, if the result shows that the reliability of heat rejection to the final heat sink is insufficient, added component or system redundancy may be chosen to improve the situation. Applied rigorously and comprehensively across the whole scope of the power plant, a PSA offers yet another dimension through which any given design can be reviewed and tested. This process is a valuable addition to the normal methodologies of engineering design review, commissioning tests, and operational review. PSA adds knowledge about the degree to which the plant is robust against a wide range of component and system failures.

10.4.2 Application to operations

The start of plant operations presents management with a new set of challenges. The operating organization is expected to operate the machine safely and productively through a plant lifetime of the order of 50 to 100 years, in other words, for up to four or five complete generations of operating staff. They are expected to retain engineering expertise as well as operating expertise through this whole time period. Fortunately there are tools available that greatly simplify this seemingly daunting task.

It is quite easy to determine the precise state of each component and system when the plant is new, provided the commissioning methodology was sufficient. In a modern plant the methodologies for handover from design, construction, and commissioning to operations includes a complete set of detailed documentation, plus a valuable electronic model of the whole plant. Such a modern computer-aided drafting and design (CADD) model (Didsbury *et al.*, 2000) gives the operating staff a final 'as built' description of the plant, down to a finely detailed description of each system, complete with a record of the history and capability of each component of that system (Petrunik and Rixin, 2003). For the first time, operations have at hand a tool for configuration management that can be used productively throughout the life of the plant.

At any given instant during operation it is possible that some components and subsystems will become unavailable. Given the complete configuration package from the electronic model it is possible to know precisely which components are unavailable. It is even possible to know this in a predictive fashion; that is, prior to start of maintenance, the staff can estimate the change in future unavailability that will be caused by this maintenance operation, and to judge its effect on the risk of continued plant operation during a planned maintenance period. Planning of maintenance is greatly simplified, and regulatory requirements for either continued operation or plant shutdown become clear and unequivocal.

Training offers the second important advantage of the comprehensive CADD model during operation. All systems and components can be 'seen' on the computer screens at any time, so that maintenance training is easier even when some area in which maintenance is required is unavailable during at-power operation. Given the long lifetime of the plant, the model presents a useful way to pass plant information from generation to generation. With necessary care and attention given to upkeep of the model, plant information becomes effectively eternal. Obviously, updates of this model can provide a means to record configuration changes that may become necessary during later plant life, such as those due to new regulatory requirements or due to a component manufacturer being replaced by a new one, and so on.

10.4.3 Limitations of probabilistic safety analysis

It is a truism that actual plant malfunctions never go 'by the book'; that is, they are always unique and do not conform to the exact sequence defined in the deterministic or the probabilistic safety analyses. Furthermore, there is never full assurance that all possible failure modes and combinations have been investigated. The most likely cause of this diversity of cause and effect is the known complexity of the plant systems, combined with the much larger complexity that arises from innate human diversity at the operating staff level. Human behaviour, both as individuals and in groups, can exert very large positive as well as negative effects on calculated frequencies and consequences. Put in another way, a highly competent operating crew can safely operate even a seriously flawed plant design; at the same time an incompetent operating crew is capable of doing great damage to even an extremely well-designed plant. Lastly, considering the long time span of plant operation (50 to 100 years), all of the important variables can range from fully satisfactory at one point in time to unsatisfactory at a later time. Human managers are always responsible for sustaining high performance (and hence low risk) at all times – and even they are never perfect. Hence, there is a basic need for audits by an independent regulatory agency.

10.5 Risk-informed decision-making processes

During the operation of a complex system such as a nuclear power plant, it is not unusual to find that one or more components have ceased to operate correctly. Given the fact that plant design incorporates extensive redundancy due to defence in depth and defence in time design and operating procedure, as described earlier in this text, there is a reasonable probability that a single failure still will leave the plant in an operable condition. That condition must be fully understood, however, to be certain that continued operation will remain safe, and will stay within the terms and conditions of the plant's operating licence. The most important benefit of this process is that it can reduce the uncertainty associated with slightly abnormal plant states. In case of uncertainty, of course, the operator is bound to return the plant to a known safe operating state.

10.5.1 Basic principles

The following is taken verbatim from the US Nuclear Regulatory Commission White Paper on risk-informed and performance-based regulation (USNRC, 1975):

> A 'risk-informed' approach to regulatory decision-making represents a philosophy whereby risk insights are considered together with other factors to establish requirements that better focus licensee and regulatory attention on design and operational issues commensurate with their importance to health and safety. A 'risk-informed' approach enhances the traditional approach by: (a) allowing explicit consideration of a broader set of potential challenges to safety, (b) providing a logical means for prioritizing these challenges based on risk significance, operating experience, and/or engineering judgment, (c) facilitating consideration of a broader set of resources to defend against these challenges, (d) explicitly identifying and quantifying sources of uncertainty in the analysis, and (e) leading to better decision-making by providing a means to test the sensitivity of the results to key assumptions. Where appropriate, a risk-informed regulatory approach can also be used to reduce unnecessary conservatism in deterministic approaches, or can be used to identify areas with insufficient conservatism and provide the bases for additional requirements or regulatory actions.

Risk-informed decision-making is an on-going activity that continues throughout the life of the plant. It is based on the safety assessment of the power plant as it exists at a given point in time, including all changes, updates, and ageing effects that are important to safety (GSR Part 4, 2009). The following is taken verbatim from the GSR document, as a statement of the necessary background for risk-based decision making:

> The responsibility for carrying out the safety assessment rests with the responsible legal person; that is, the person or organization responsible for the facility or activity – generally, the person or organization authorized (licensed or registered) to operate the facility or to conduct the activity. The operating organization is responsible for the way in which the safety assessment is carried out and for the quality of the results. If the operating organization changes, the responsibility for the safety assessment has to be transferred to the new operating organization. The safety assessment has to be carried out by a team of suitably qualified and experienced people who are knowledgeable about all aspects of safety assessment and analysis that are applicable to the particular facility or activity concerned.

Clearly, the operating organization is expected to establish the infrastructure for carrying out the safety assessment to the satisfaction of the national regulatory agency. In addition, it is essential to find a proper framework for carrying out a satisfactory decision-making process to enable risk-informed decisions.

10.5.2 Application during plant operation

Support groups in operating organizations have established so-called 'living PSA' analysis systems with responsibility for daily updating of the plant operating and maintenance state, forward planning of scheduled maintenance operations, and contingency planning programs to predict the correct course of action for shift management personnel in the event of anticipated abnormal operating states. These incremental risk estimates are based on the latest updated version of the plant safety assessment.

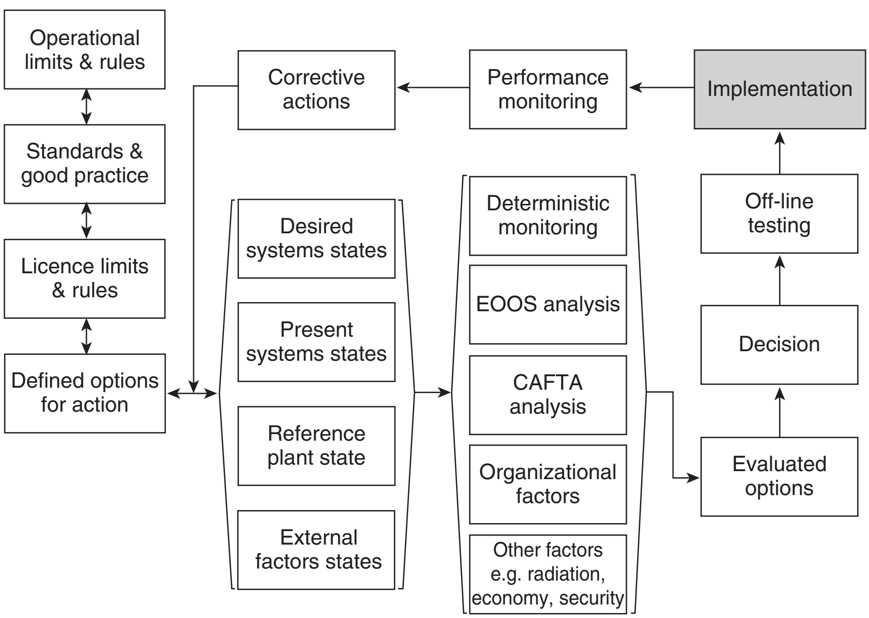
A good example of this application to power plant operation is the software package named EOOS (Equipment Out Of Service), the risk and reliability workstation (EOOS Demo 3.5, 2008) produced by the Electric Power Research Institute (EPRI). EOOS is independent of other EPRI reliability analysis software such as the Cutset and Fault Tree Analysis (CAFTA) system (CAFTA, 2009) but uses many of the same conventions.

> EOOS uses a safety or risk model of the plant, based on fault tress and minimal cutsets, such as those developed in a Probabilistic Risk Assessment (PRA). EOOS wraps a user-friendly interface around these reliability analysis tools to make them accessible to non-PSA experts.
>
> EOOS communicates in the language of its users – using the familiar terminology of components, trains, systems, tests, and clearances. Using the current plant configuration, EOOS can propagate information through the model and quantify risk measures. EOOS translates fault tree results into color-coded status panels, timelines, and lists of relevant and risk-significant activities. Within seconds, an EOOS user can identify a safety problem, and the specific work activities that cause it. The EOOS user will then have the information to decide whether the problem is significant enough to warrant special contingency actions.
>
> *The software offers various benefits and values for the user. EOOS can help reduce Operation and Maintenance (O&M) costs by: (a) reducing the chance of a costly operational mistake.* As unplanned events creep into a well-planned work schedule, you run the risk of unexpected reductions in plant safety. EOOS detects these safety problems that routinely escape the scrutiny of safety reviews based on train-level work windows, (b) by *reducing the labor needed to perform safety reviews*. An EOOS model integrates the safety impact of *all* work tasks affecting *all* risk significant safety functions into concise screen presentations and printed reports, (c) by p*roviding credible, risk-based insights that minimize unnecessarily conservative requirements*. EOOS results can become the basis for eliminating requirements that increase outage duration, without a commensurate safety benefit.

10.5.3 Integrated risk-informed decision-making process

The International Nuclear Safety Advisory Group has prepared a draft proposal for an integrated process, as described in INSAG-25 (2010). This document aims at the most difficult of all safety-related actions; that is, the decision process surrounding the question of what is a sufficient level of safety. As noted in the INSAG document, the process must be flexible to adapt to the myriad of different situations under which these decisions must be made. For example, the decision process to be applied during the stage of conceptual design of a plant can be much broader and more thoroughly researched than can the process that must be applied when (purely for example) a redundant pump fails for some reason during operation and the appropriate subsequent action must be decided. The difference lies mainly in the time available for decision and action – much shorter in the second case.

The general roles of the major stakeholders in the safety management system were discussed in Section 10.2.4. Specific relationships during plant operation are much more complex, but at a higher level always consist of an operating organization overseen and audited by experienced and independent technical staff on behalf of the licensing authority. Fig. 10.8



10.8 An integrated risk-informed decision-making concept.

illustrates a somewhat more detailed map of one possible set of processes involved in arriving at a safety-related decision.

To alleviate the enormous expenditure of resources involved in a step-by-step operation of the process involved in Fig. 10.8, it is usual to develop a set of symptom-based operating procedures for use by the individuals and groups who actually operate the plant controls. Properly developed and tested, these procedures can dramatically shorten the decision and action time required. Procedures must be developed through a comprehensive and interactive process such as that described in INSAG-25 (2010). They must also be periodically updated based on operating experience. The same rule must hold for equipment or design modifications that may be required periodically in a mature operating plant. Finally, if significant changes occur in the external environment of the plant (for example, a newly discovered threat) then a review may result in changes to operating procedures or equipment in order to deal with the new situation.

10.6 Impact of past accidents on future safety improvement

This section could be titled 'learning from experience'. This issue is somewhat broader, however, as necessitated by a complex system such as nuclear energy. This issue becomes extremely important to support the justification of a new nuclear program, because the experience must be gained rapidly, essentially without the benefit of prior operating experience by the operating company. It has been found very useful to study abnormal events that have occurred in nuclear power plants in the past. By this means the newly initiated operating staff get a clear picture of how mistakes have been made, how consequences of those mistakes have been dealt with, and how future operation can benefit from the lessons learned (Duffey and Saull, 2003, 2008). Further, it has been found that the causes, patterns and frequencies of failure are very similar across a wide range of human endeavour. The nuclear energy enterprise stands out (Weick and Sutcliffe, 2007) as a high-performance industry in terms of its low risk of high-consequence accidents.

10.6.1 Exchange of operating experience

National and international organizations have been established to assist with new program startup. Most work of the Division of Reactor Engineering at the IAEA is dedicated to education and communication between the power programs of member states. In addition, the World Nuclear

Association comprises mainly companies. Current members are responsible for virtually all of world uranium conversion and enrichment production and some 85% of world nuclear generation. Further, the World Association of Nuclear Operators (WANO) has the mission to 'Maximize the safety and reliability of nuclear power plants worldwide by working together to assess, benchmark, and improve performance through mutual support, exchange of information, and emulation of best practices.' The WANO organization grew out of the US-based Institute for Nuclear Power Operations (INPO) that was established shortly after the Three Mile Island accident in 1979.

In addition to these broad-based organizations, a number of plant-specific owners' groups operate around the world. As an example, the CANDU owners group (COG) is a 'not-for-profit organization dedicated to providing programs for cooperation, mutual assistance and exchange of information for the successful support, development, operation, maintenance and economics of CANDU technology.' All operators of CANDU plants worldwide are members of COG. Together, these organizations provide major support for any new member, ranging from general education on aspects of this technology, through specific training for operating staff, posting of individuals to operating nuclear units, and cooperative R&D to maintain and improve operating stations. The overall effect is to reduce the operating cost of each plant. Essentially all vendors of nuclear stations have established similar organizations in order to assist operational organizations to maintain modern understanding of their facilities.

10.6.2 Learning from accidents

This aspect of learning is not very different from the exchange of information on normal operation as discussed in the previous section. Normal operation also includes a host of small equipment malfunctions and human errors – all of which are examined to find out if they might be precursors of larger malfunctions that could occur in the future.

We must carefully define the usage of the word 'accident' in this context, beyond the conventional usage. We are dealing here with a complex technology for which all contingencies are presumed to be subject to careful engineering analysis and design. It is reasonable, therefore, to take the position that *all* unfortunate consequences arise from human error at some stage of the process. This classification is somewhat at odds with usual practice; however, taking the example of an equipment failure, one can quickly identify different causes – design error, manufacturing error, installation error, and maintenance error. *All* of these failures are caused by human failure. Even so-called natural events are expected to be protected against by design (through either prevention or mitigation).

An example

Given the fact that accidents are, at the very least, caused mostly by human error, it is very useful (Duffey and Saull, 2008) to look at serious accidents that occur in other industries and human activities in general. The reason for this is to 'normalize' the accident rate in nuclear plants to the usual patterns of human existence. Indeed, Reason (1990) points out that 'active' human errors are very rare in the world nuclear industry when compared with the frequency of correct action.

Table 10.1 outlines a 'typical' accident sequence. (Note that the specific technology is of secondary importance in this type of analysis.) In this case a sudden tire failure led to failure of one engine during takeoff. The pilot was 1.5 seconds late in applying the takeoff abort procedure, and so the immediate cause of the accident was said to be pilot error. During subsequent review and analysis it was found that a number of other factors actually had a powerful negative effect on the accident – most especially the continued use of tires that were already beyond their service life. In the end, it became apparent that airline management was strongly implicated through unsafe practices.

Another example

The Challenger space orbiter failure in 1997 and the Columbia failure in 2003 each displayed several contributing elements, but the root cause in both cases was human error. Risks were taken without full understanding of the probabilities and without proper balance in senior management decision making – as judged *post facto*. It is interesting that in none of these cases was the future risk of the event recognized before the fact, even

Table 10.1 A typical accident – Toronto Airport

DC 9 initiated a takeoff run	Everything appeared normal up to speed V1
Sudden loss of power from one engine	After V1, aircraft cannot be stopped safely – takeoff is mandatory
Pilot hesitates for 1½ seconds before applying 'abort' procedure	Abort procedure: full reverse engine thrust and brakes, warn passengers to brace
Pilot steers to left to avoid runway light standards	He knows the aircraft will overshoot the runway
Aircraft coasts off apron and glides into a ravine	The aircraft was below stall speed when it entered the ravine
Fuselage breaks in half, killing two passengers	All other passengers and crew escaped

though more junior staff gave clear warnings in all three cases. (The same pattern existed before the recent oil drilling disaster in the Gulf of Mexico.)

A general principle

Risk experts (Mullane, 2006) have identified a pattern of human response that helps to explain many similar accident events; it is called the 'normalization of deviation'. Looking back at the Toronto Airport case, suppose that the practice of using older tires beyond their service life had succeeded in the past. Since the apparent result showed better airline economics, the practice would be encouraging to management; it would become the *normal* practice. Certainly, this pattern emerged in the case of the Challenger booster rockets. Previous launches had succeeded even though the O-ring seals had leaked – the practice of launching with off-normal seals had become *normal*. The same behaviour pattern existed in the case of the Columbia external fuel tank insulation failure. Insulation had fallen off the tank during launch several times and had sometimes hit the orbiter, but the mission still succeeded. Observing insulation loss during launch had become *normal*. It will not be surprising if this same pattern emerges from the Deepwater Horizon investigation when that is completed. There are many other earlier examples that could be cited.

10.6.3 Institutional failure

David Mosey (Mosey, 2006) has examined several cases from the short history of nuclear energy. In the second edition this author cites four interrelated types of management error under four general headings:

1. Misperception of hazard. Lack of accurate and consistent understanding of the specific demands/vulnerabilities of the technology.
2. Dominating production imperative. Production considerations override safety. Safety is under-resourced.
3. Unassigned/undefined safety responsibility/authority. Failure to assign, define or assume safety responsibility and/or authority completely or clearly.
4. Failure to recognize, acknowledge or respond effectively to an unsatisfactory or deteriorating safety situation. 'Denial' or 'unawareness', or the failure to learn from experience, is included here.

Error number 4 includes, of course, the category of 'normalization of deviation' discussed earlier.

David Mosey clearly illustrates the importance of safety culture to the successful long-term operation of complex technologies. Quite obviously, senior management has a powerful influence on the performance of the whole organization within their authority. Less obvious, and often a neglected factor, is the influence that should be exerted 'from the bottom up'. To say this in another way, the knowledge flow from junior to senior ranks must be fostered and encouraged. Senior management must be knowledgeable of the details of the organization they manage. This requirement is *opposite* to the older notion, promulgated by some business schools, that the quality of a manager could be considered independent of the specific activities of the managed organization.

10.7 Evolution of major safety performance indices

Over the past half-century there have been important developments in the measurement of safety of complex technologies, notably aircraft and nuclear safety (Duffey and Saull, 2003; Reason, 1990; Weick and Sutcliffe, 2007). Safety (or its logical inverse, risk) is difficult to measure when it is good; that is, when nothing happens by which to measure the frequency of abnormal occurrences. Under these conditions it is natural for humans to conclude that the risk is very low or zero. The second level worthy of consideration is the frequency of 'close calls', or situations that could have resulted in negative consequences had some fortuitous occurrence not intervened. A 'close call' is a clear indication of loss of defence in depth, within the safety regime applied to nuclear plant operation. A third level of defence is available through examination of the availability of 'poised' or operation-ready safety systems designed to mitigate the consequences of abnormal events. All of these performance indices rely on administrative attention and action by management and by independent safety auditors assigned to ensure that safety-significant events are actually observed and recorded.

10.7.1 Normal operation

The first level of safety is always to be found in the education and training of all those involved in the nuclear energy enterprise, from designers to senior management and finally to junior operating staff. The concept of safety culture (INSAG-4, 1991) is carefully fostered in the industry, to build and sustain the habits of management and job execution that are known to support safe operation of the system.

Codes of good practice

Programs administered by owners' groups and national/international organizations such as INPO and WANO are very active in developing codes of good practice for promulgation across the world. These codes represent the best judgment of true experts in the field of nuclear plant operation. These are made available to all members of these organizations; they offer sound support to any newly founded operating organization.

Regulatory staff posting to operating stations

It has been found strongly beneficial to plant operational safety to assign a small staff of regulatory personnel to each operating station. This practice keeps the regulatory agency abreast of the latest technical and managerial information, and provides plant operations staff with immediate feedback of the opinion of the regulator to any continuing or novel situation at the plant. This field staff is, of course, supported by the central technical groups, usually posted to the headquarters of the regulatory agency.

Periodic testing of safety-related systems

All components and systems that are important to safety are tested at regular intervals, with the time between tests depending on the specific characteristics of the component. (In practice, essentially all plant systems are important to safety to some degree.) These data are added to the existing probabilistic safety assessment model to keep it up to date and to provide a current estimate of the component, subsystem, and overall plant systems reliability to respond to operational and safety demands. In a very real sense, the testing and maintenance activities help to keep the whole plant in 'good as new' condition over its whole operating life.

Special safety systems availability

In the older Canadian licensing tradition, three special safety functions were designated – shut down, close the containment boundary, and cool the fuel. (These functions are universally recognized in international documents and practice.) The unavailability of each system was required to be less than 10^{-3} per demand. Recognizing the primary importance of reactor shutdown after some abnormal occurrence, two independent shutdown systems were required after the first commercial 43-unit plant was installed at Pickering. Regular testing of each of the special safety systems was required during plant operation; test results were reported to the regulatory agency in order to ensure that each unavailability requirement was being met. (In practice they were not always met; subsequent effort then immediately became an action item on the part of the operating organization.)

Safety support systems

Obviously, special safety systems may require support services such as status monitoring, control signals, power supplies, water supplies, pump and valve operating power so that they can carry out their designated function. This is especially true of post-accident fuel and containment cooling that may be required to operate for months in some accident situations. Regular reporting of operational testing and maintenance to the regulatory authority is an integral part of the test program – essential for the staff to carry out their central auditing function.

10.7.2 Abnormal operating events

So-called significant event reports are filed for each off-normal situation encountered during operation. These reports are filed in two different forms; the first identifies all staff involved in the event and their role in either initiating or mitigating the abnormality being reported. This report is used *only* by plant management for performance reporting and, if necessary, for retraining or discipline of individual staff members. The second form of a significant event report is distributed to regulatory staff, other operating plants, and design groups to serve as a detailed record of abnormal events that may provide lessons for improvement of future operating procedures. Abnormal event reports are distributed widely in order to maximize the benefits of the specific learning experience as well as contributing to nuclear station operating experience in general. These events are analyzed by operations support groups around the world, and appropriate actions are taken. A condensed version of the events deemed to be most important to the broader community is forwarded to the IAEA incident reporting system (INIS) as well as to other international organizations.

Generic action items

This important classification of significant events is meant to serve as information (usually regulatory issues) on which action must be taken, either design, analysis, and/or research and development – to resolve outstanding *newly identified* issues that arise in operation. These issues are resolved, normally, through collaborative work between members of the plant owners groups. They are reviewed, analyzed, and eventually disposed by the national regulatory staff.

10.8 Sources of further information and advice

In order to establish a new nuclear energy program, a country must, of course, first establish a sound knowledge base so that decisions about the direction to be taken are sound and in the interest of the nation concerned. Even with all possible goodwill on the part of outside organizations, they are very unlikely to fully understand the goals of any nation as well as do its native inhabitants. Fortunately, the number of channels of communication is vast, and opportunities for education of staff are excellent.

10.8.1 Open literature

The world inventory of available published literature already contains much of the history and technology of the nuclear energy enterprise over the past 60 years. Many conference proceedings, reports and textbooks are freely available in libraries, a few of which are listed here. Naturally, some information is restricted for reasons of commercial interest.

10.8.2 Owners' groups

Owners' groups mentioned in Section 10.6.1 are sharply focused on sustaining good performance of their own power plants. These groups encourage joint R&D and education of operating staff. For example, the CANDU group website can be found at COG (2010). Generally, this site offers information to the owners of CANDU power plants; other examples are AREVA-NP (2010), General Electric (2010), and the Westinghouse Owners' Group (WOG; unfortunately, no reference available). One general characteristic of these groups is that they maintain all or some of their information confidential to group members. This is understandable due to the large commercial interests involved.

10.8.3 Institute for Nuclear Power Operations (INPO) and international organizations

The INPO organization was formed in 1979 in the wake of the Three Mile Island nuclear plant accident. A number of US industry leaders recognized that the industry must do a better job of policing itself to ensure that an event of this magnitude should never happen again. INPO was formed to establish standards of excellence against which the plants are measured. An inspection of each member plant is typically performed every 18–24 months. The Institute's programs include:

- SEE-IN (an information sharing network)
- EPIX (an equipment failure database)
- National Academy for Nuclear Training
- Events analysis
- Human performance
- Accreditation
- Evaluations.

Information regarding INPO as well as the Nuclear Energy Institute, the International Atomic Energy Agency, and the World Association of Nuclear Operators can be found at WANO (2010).

10.8.4 World Nuclear Association (WNA)

The World Nuclear Association (WNA, 2010) is the international organization that promotes nuclear energy and supports the many companies that comprise the global nuclear industry.

WNA arose on the foundations of the Uranium Institute (UI) established in London in 1975 as a forum on the market for nuclear fuel. In 2001, spurred by the expanding prospects for nuclear power, the UI changed its name and mandated itself to build a wider membership and a greater diversity of activities. The goal was to develop a truly global organization geared to perform a full range of international roles to support the nuclear industry in fulfilling its enormous growth potential in the twenty-first century.

Since WNA's creation in 2001, the effort to build and diversify has born fruit. WNA membership has expanded three-fold to encompass (1) virtually all world uranium mining, conversion, enrichment and fuel fabrication; (2) all reactor vendors; (3) major nuclear engineering, construction, and waste management companies; and (4) nearly 90% of world nuclear generation. Other WNA members provide international services in nuclear transport, law, insurance, brokerage, industry analysis and finance. WNA will remain a work in progress. Its rapid growth reflects recognized value and represents major advance in building toward universal industry membership. Today WNA serves its membership, and the world nuclear industry as a whole, through actions to:

- Provide a global forum for sharing knowledge and insight on evolving industry developments
- Strengthen industry operational capabilities by advancing best-practice internationally
- Speak authoritatively for the nuclear industry in key international forums
- Improve the international policy and public environment in which the industry operates.

10.9 References

AREVA-NP (2010), PWR Owners' Group website, available from http://www.us.areva-np.com/enewsletters/TheSource/The.Source_Vol.IV_no.04.html (accessed 22 June 2010)

CAFTA (2009), CAFTA, FRANX, EOOS description and contact information at http://teams.epri.com/RR/Art/_w/CAFTA_FTREX_card_2009_FINAL_Page_2_jpg.jpg

CNSC (2000), Laws and Regulations, available from http://www.cnsc-ccsn.gc.ca/eng/lawsregs/ (accessed 22 June 2010)

COG (2010), CANDU Owners' Group website, available from http://www.candu.org/, Toronto, CANDU Owners Group Inc. (accessed 22 June 2010)

Didsbury R, Shalaby BA, and Torgerson DF (2000), *The Application of an Integrated Approach to Design, Procurement, and Construction in Reducing Overall Nuclear Power Plant Costs*, Sheridan Park, Mississauga, Ontario, AECL

Duffey RB and Saull JW (2003), *Know the Risk: Learning from Errors and Accidents: Safety and Risk in Today's Technology*, Boston, MA, Butterworth-Heinemann

Duffey RB and Saull JW (2008), *Managing Risk, The Human Element*, Chichester, UK, John Wiley & Sons

EOOS Demo 3.5 (2008), *EOOS, Software Manual,* EPRI, Palo Alto, CA

Farmer FR (1967), *Siting Criteria – A New Approach*, Paper No. SM-89/34, IAEA Conference on Containment and Siting, Vienna, IAEA

General Electric (2010), BWR Owners' Group website, available from http://www.gepower.com/prod_serv/products/nuclear/en/bwr_owners_group/index.htm (accessed 22 June 2010)

GSR Part 4 (2009), *Safety Assessment for Facilities and Activities: General Safety Requirements*, STI/PUB/1375, Vienna, IAEA

Hurst DG and Boyd FC (1972), http/canteach.candu.org/library/20051707.pdf – powered by Google Docs.webarchive

IAEA (2006), *Fundamental Safety Principles,* Safety Fundamentals No. SF-1, Vienna, IAEA

INSAG (1988), *Basic Safety Principles for Nuclear Power Plants*, 75-INSAG-3, Vienna, IAEA

INSAG (1991), *Safety Culture*, 75-INSAG-4, Vienna, IAEA

INSAG (1992), *Probabilistic Safety Assessment*, 75-INSAG-6, Vienna, IAEA

INSAG (1995a), *A Common Basis for Judging the Safety of Nuclear Power Plants Built to Earlier Standards*, INSAG-8, Vienna, IAEA

INSAG (1995b), *Potential Exposure in Nuclear Safety*, INSAG-9, Vienna, IAEA

INSAG (1996), *Defence in Depth in Nuclear Safety*, INSAG-10, Vienna, IAEA

INSAG (1999a), *Basic Safety Principles for Nuclear Power Plants*, 75-INSAG-3, Rev. 1, INSAG-12, p. 15, Vienna, IAEA

INSAG (1999b), *Management of Operational Safety in Nuclear Power Plants*, INSAG-13, Vienna, IAEA

INSAG (2002), *Key Practical Issues in Strengthening Safety Culture*, INSAG-15, Vienna, IAEA

INSAG (2003), *Maintaining the Design Integrity of Nuclear Installations Throughout Their Design Life*, INSAG-19, Vienna, IAEA

INSAG (2006), *Stakeholder Involvement in Nuclear Issues*, INSAG-20, Vienna, IAEA

INSAG (2010), *A Framework for Integrated Risk-Informed Decision Making Process*, INSAG-25, Vienna, IAEA

Meneley DA (1999), *Risk Analysis Methods*, available from University of Ontario Institute of Technology, Course ENGR 4660U, Oshawa, Ontario

Mosey D (2006), *Reactor Accidents, Institutional Failure in the Nuclear Industry*, second edition, Sidcup, UK, Nuclear Engineering International

Mullane M (2006), *Riding Rockets: The Outrageous Tales of a Space Shuttle Astronaut*, New York, Scribner

Murley TE, Rosztoczy ZR, and McPherson GD (1991), The evolution of the structure and application of US NRC regulations and standards, *Nuclear Engineering and Design*, Volume 127, Issue 2, 219–224

NS-R-1 (2000 and 13 associated guides), *Safety of Nuclear Power Plants: Design: Safety Requirements*, STI/PUB/1099, Vienna, IAEA

NS-R-4 (2005 and 6 associated guides), *Safety of Research Reactors: Safety Requirements*, STI/PUB/1220, Vienna, IAEA

Petrunik KJ and Rixin K (2003), *CANDU Project Construction Experiences and Lessons Learned to Reduce Capital Cost and Schedule Based on Qinshan CANDU Project in China*, available from http://canteach.candu.org/library/20031701.pdf (Accessed 22 June 2010)

Rasmussen N *et al.* (1975), *The Reactor Safety Study*, Washington DC, USNRC

Reason J (1990), *Human Error*, Cambridge, Cambridge University Press

Snell VG (2001), *CANDU Safety #22 – Regulatory Requirements for Design*, available from http://canteach.candu.org/library/19990123.pdf (accessed 22 June 2010)

USNRC (1975), Executive summary: main report (PWR and BWR), *Reactor Safety Study: An assessment of accident risks in US commercial nuclear power plants*, Rockville, MD, USNRC

USNRC (1991), *Requirements for monitoring the effectiveness of maintenance at nuclear power plants*, 10 CFR 50.65, Washington, DC, USNRC

USNRC (2010), *General design criteria for nuclear power plants*, Appendix A to 10 CFR Part 50, Washington, DC, USNRC

WANO (2010), World Association of Nuclear Operators website, available from http://www.nucleartourist.com/basics/inpo.htm (accessed 22 June 2010)

Weick KE and Sutcliffe KM (2007), *Managing the Unexpected: Resilient Performance in an Age of Uncertainty*, Second Edition, San Francisco, Wiley

WNA (2010), World Nuclear Association website, available from http://www.world-nuclear.org/ (accessed 22 June 2010)

11
Radiation protection in nuclear power programmes

A. J. GONZÁLEZ, Autoridad Regulatoria Nuclear de Argentina (Argentine Nuclear Regulatory Authority), Argentina

Abstract: The international radiation protection system for nuclear power plants (NPPs) is described. It includes the estimates of the United Nations Scientific Committee on the Effects of Atomic Radiation, the recommendations from the International Commission on Radiological Protection, and the safety standards established by the International Atomic Energy Agency. The aim is to summarize the sytem's fundamental principles. Their application to potential exposures (and therefore to nuclear safety) is portrayed including a compliance criterion for prospective probabilistic safety assessments. Practical considerations on occupational and public protection are discussed, including a description of the latest assessments of the radiological consequences of the Chernobyl accident.

Key words: radiation safety, radiation protection, nuclear safety, safety assessment, nuclear regulation.

11.1 Introduction

Ionizing radiation (named, in short, *radiation*) is perceived to be the nemesis of nuclear energy. This is unsurprising: radiation exposure is detrimental to human health and omnipresent in activities and installations of the nuclear fuel cycle, including regulated *nuclear power plants* (hereinafter termed *NPPs*) for electricity production. These installations routinely discharge into the atmosphere and watercourses, gases, aerosols and liquids containing small amounts of radioactive substances, which may cause radiation exposure to members of the public; their operators are occupationally exposed to radiation delivered by ubiquitous radioactive materials in workplaces. NPP safety assessments demonstrate that the likelihood of a catastrophic accident is exceedingly small; however, should a nuclear accident occur its consequences can be severe: emergency workers may be exposed to high radiation levels and large amounts of radioactive materials can be uncontrollably released into the environment, contaminating vast territories and exposing large populations. NPPs also generate large amounts of radioactive waste that have to be transported over public places and which are viewed as a radiation exposure legacy for our descendants.

Decommissioning activities, necessary for the termination of nuclear operations, may leave radioactive residues that will likely remain in the habitat. Ultimately, concerns have been growing on the security of the radioactive materials in the nuclear fuel cycle since their malevolent use might cause serious radiological harm.

Predictably, the protection against radiation exposure, namely *radiation protection* (sometimes termed *radiological protection*), has become a *sine-qua-non* condition for the justification of nuclear power.

The protection against radiation exposure has been fully internationalized and the current radiation protection rests on four international foundations:

1. The *estimates of radiation levels and effects* are assessed by the United Nations Scientific Committee on the Effects of Atomic Radiation (UNSCEAR). UNSCEAR is an intergovernmental scientific body founded in 1955 and since reporting radiation levels and effects to the United Nations General Assembly (UNGA) (UNSCEAR, 1958, 1962, 1964, 1966, 1969, 1972, 1977, 1982, 1986, 1988, 1993, 1994, 1996, 2000, 2001, 2009, 2011).
2. A *radiation protection paradigm* is recommended by the International Commission on Radiological Protection (ICRP). ICRP is a scientific non-governmental independent charity, i.e., a non-profit-making organization, providing advice on radiation protection (ICRP, 1951, 1955, 1957, 1959, 1964, 1966, 1977, 1978, 1985a, 1985b, 1991, 2007a). It was established in 1928 by the International Congress of Radiology, with the name of the International X-Ray and Radium Protection Committee (IXRPC) (IXRPC, 1928, 1934), following a decision by the Second International Congress of Radiology, and in 1950 it was restructured and renamed as now.
3. *International standards on radiation safety* are established under the aegis of the International Atomic Energy Agency (IAEA) (IAEA, 1960, 1962, 1967, 1976, 1982, 1996a, 2011), lately in co-sponsorship with other relevant intergovernmental organizations within the United Nations (UN) system, therefore becoming the *de facto* international radiation protection authority. Since its creation in 1957, the IAEA has been responsible for safety-related functions that are precisely described in its Statute, namely (1) establishing standards of safety for the protection of health against the detrimental effects attributable to radiation exposure; and (2) providing for the application of those standards at the request of any State.
4. *Global provisions for the implementation of radiation safety standards*, through mechanisms put in place by national agencies and by the IAEA and other international organizations.

On the basis described heretofore, this chapter will explore the international approach for radiation protection at nuclear activities in general and the nuclear fuel cycle and its NPPs in particular. Its aim is to provide guidance on the fundamental principles on which appropriate radiation protection can be based rather than a regulatory text. International radiation protection trends and achievements have been reviewed at the recent 12th Congress of the International Radiation Protection Association (IRPA): Strengthening Radiation Protection Worldwide: Highlights, Global Perspective and Future Trends (IAEA, 2010; González, 2009a).

The chapter will not review a number of issues closely related to radiation protection and NPPs, *inter alia* the following:

- The radiological security of radiation sources at NPPs and of the NPP itself, a subject for which there are detailed recommendations from ICRP (ICRP, 2005a) and which has been amply reviewed by the IAEA (IAEA, 1999b, 2000a, 2001, 2003a, 2006c) and by the author (González, 1999a, 1999b, 2001b, 2003a, 2006)
- The radiation protection aspects of waste and spent fuel management, an issue that is discussed separately in Chapter 14, for which there are several recommendations from ICRP (ICRP, 1985b, 1997a, 1998) and which has been thoroughly discussed globally, mainly at the IAEA (IAEA, 2003d; González, 2000, 2003c), and is regulated by an international convention (IAEA, 1997)
- The radiation safety of the transport of nuclear and other radioactive materials associated with nuclear fuel cycle operations, an activity that is heavily regulated globally by standards (IAEA, 2008b, 2009) constituting a real international regime (González, 2004a) and for which there is a global consensus (IAEA, 2004a)
- The radiation legacy from the termination of NPP operation and the consequent decommissioning, and also from accidents, a subject to be discussed in Chapter 24, which has been the subject of intense IAEA activity (IAEA, 2003c, 2007) and ICRP recommendations (ICRP, 2009a) but which still lacks an international regime (González, 2003d)
- Last but not least, the radiological consequences of NPP accidents (except the Chernobyl accident, which will be briefly covered hereinafter), and the protection of people in emergencies, a subject covered by ICRP (ICRP, 2009b), as well as emergency planning and preparedness, a subject that will be treated in Chapter 12.

11.2 Radiation doses

11.2.1 Quantities and units

A unique characteristic of radiation protection is the full international harmonization of the relevant quantities and units. This has been achieved under the influence of the International Commission of Radiation Measurements and Units (ICRU), a sister organization to ICRP (ICRU, 1938, 1954, 1962), and is unique *vis-à-vis* other pollutants.

NPPs are characterized by the presence of radioactive substances, the amount of which is described by the quantity termed *activity* and measured in the unit termed *becquerel* – although, in the past, the unit *curie* was and still is widely used. One becquerel represents an extremely small activity; for instance, one becquerel is the activity of potassium (which is a long-lived, naturally radioactive element) contained in less than one-tenth of one banana! (Conversely, 1 curie represents a significant activity as it equates to 37 thousand million becquerels.) Although varying among plants, NPPs currently discharge into the environment an average activity of around a hundred million million becquerels (or terabecquerels, TBq) per kilowatt year of electrical energy produced, mainly of short-lived radioactive noble gases.

NPP materials with activity emit radiation that may expose both workers and members of the public and may be delivered from outside the person's body (*external exposure*) or by radioactive substances arising from those materials that may be incorporated into the body via inhalation or ingestion, or through open wounds or the skin (*internal exposure*). The potential health consequences on people caused by their exposure depend on the amount of radiation received, and also on the types of radiation involved and the organs exposed.

The amount of radiation is measured in terms of the quantity termed the *radiation dose*, and that received by human tissues is termed the *absorbed dose* and is assessed in units called *grays* (in the past, the unit *rad* was used) or in its sub-multiple, the *milligray*. One milligray is approximately the lowest annual dose absorbed by a human being due to exposure to natural background radiation.

Different types of radiation have different effectiveness to induce damage and, therefore, the absorbed dose has to be weighted by *radiation weighting factors*, w_r, to take into account the effectiveness of various radiation types. The resulting weighted quantity is termed the *equivalent dose*.

Table 11.1 (ICRP, 2007a) presents the currently recommended radiation weighting factors, w_R, which gives a general idea of the radio-efficiency of the various radiation types. It is noted that the main radiation types in NPP exposures, which are γ rays, i.e. photons, and β particles, i.e., electrons, have a weighting factor equal to 1. The weighting factor for neutrons can be high depending on their energy, but exposure to neutrons is important for some NPP equipment but normally not for people.

Similarly, different organs and tissues have different sensitivity to radiation exposure, and therefore the equivalent dose has to be weighted by *tissue weighting factors*, w_T, to take into account the various sensitivities to radiation of various organs and tissues. The quantity resulting from

Table 11.1 Recommended radiation weighting factors for calculating equivalent dose

Radiation type	Radiation weighting factor, w_R
Photons	1
Electrons[a] and muons	1
Protons and charged pions	2
Alpha particles, fission fragments, heavy ions	20
Neutrons	$W_R = \begin{cases} 2.5 + 18.2e^{-[\ln(E_n)]/6}, & E_n < 1\,\text{MeV} \\ 5.0 + 17.0e^{-[\ln(2E_n)]^2/6}, & 1\,\text{MeV} \leqslant E_n \leqslant 50\,\text{MeV} \\ 2.5 + 3.25e^{-[\ln(0.04E_n)]^2/6}, & E_n > 50\,\text{MeV} \end{cases}$

[a] Auger electrons require special treatment.

Table 11.2 Recommended tissue weighting factors for calculating effective dose

Tissue	w_T	$\sum w_T$
Bone-marrow (red), colon, lung, stomach, breast, remainder tissues[a]	0.12	0.72
Gonads	0.08	0.08
Bladder, esophagus, liver, thyroid	0.04	0.16
Bone surface, brain, salivary glands, skin	0.01	0.04
Total		1.00

[a] Remainder tissues: adrenals, extrathoracic (ET) region, gall bladder, heart, kidneys, lymphatic nodes, muscle, oral mucosa, pancreas, prostate (♂), small intestine, spleen, thymus, uterus/cervix (♀).

weighting the equivalent dose is termed the *effective dose*. Table 11.2 (ICRP, 2007a) presents the currently recommended tissue weighting factors, w_T, which gives a general idea of the radio-sensitivity of the various tissues.

Both *equivalent dose* and *effective dose* are measured in a unit termed the *sievert* (in the past, the unit *rem* was used) and in its sub-multiple the *millisievert*, which is equal to a thousandth of a sievert and to 100 thousandths of 1 rem, or 100 millirem. For all practical purposes, in most exposure situations at NPPs, one millisieviert of effective dose is quasi-equivalent to one milligray of absorbed dose.

The equivalent dose, which is used to express tissue and organ doses, and the effective dose, which is used for assessing the whole body implications, can only be formally used for normal radiation protection purposes, i.e. for situations causing relatively low doses, and cannot be properly used to express high doses that may be incurred for instance in an accident; in these cases the absorbed dose in gray or milligray should be used. Radiation protection standards usually contain universally agreed *nominal* coefficients or factors for converting activity and particle fluence into absorbed dose and also equivalent dose and effective dose.

Figure 11.1 (adapted from ICRP, 2007a) illustrates the interrelation among the radiation protection quantities, including the nominal conversion factors.

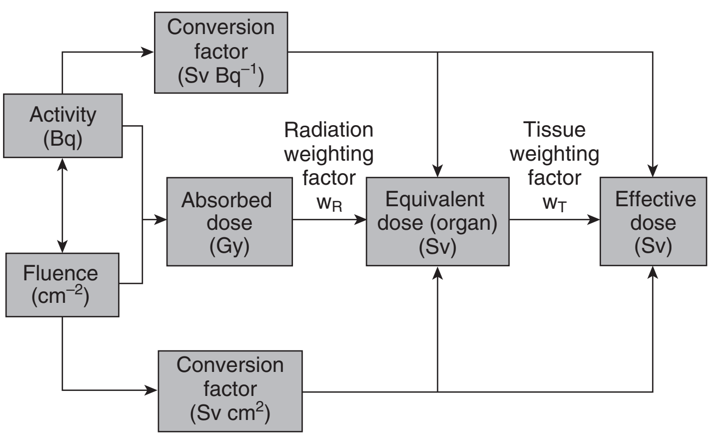
The radiation protection quantities are not directly measurable. Instruments for assessing doses in people or in the ambient environment are usually calibrated against physical operational quantities rigorously defined by ICRU and incorporated in international standards. These are the *personal dose equivalent* and the *ambient dose equivalent*. They are also expressed in sieverts and, numerically, they approximately correspond to the radiation protection quantities. The operational quantities are formally used at NPPs for verification of compliance with standards.

For reasons of simplification, this chapter will use the term dose to mean generally and indistinctly any dose quantity and will express this quantity mainly in the unit millisievert (mSv).

11.2.2 Levels

Exposure of the public

Public exposure to radiation arises not only from NPPs, but from natural background radiation and artificial sources such as medical diagnostic and therapeutic procedures, nuclear weapons testing, and many occupations that enhance exposure to artificial or natural radiation. All these sources



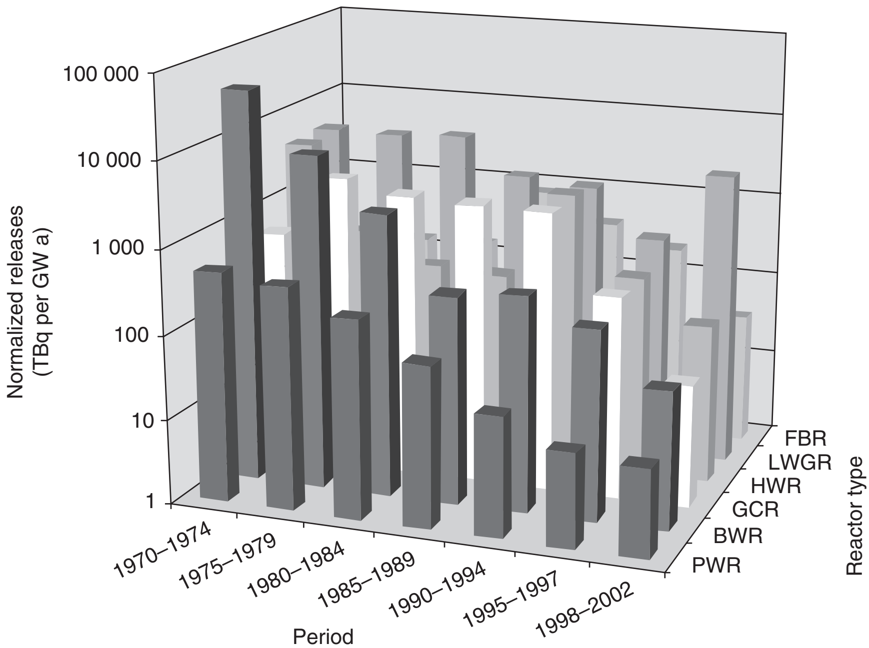
11.1 Interrelation among the radiation protection quantities.

deliver doses to members of the public, which are routinely estimated by UNSCEAR.

For as long as they have been on the planet, humans have been exposed to radiation from natural sources, and such exposure has been continually modified by human activities. The main natural sources of exposure are cosmic radiation and natural radionuclides found in the soil and in rocks. Cosmic radiation is significantly higher at the cruising altitudes of jet aircraft than on the Earth's surface. External exposure rates due to natural radionuclides vary considerably from place to place, and can range up to 100 times the average. An important radionuclide is radon, a gas that is formed during the decay of natural uranium in the soil and that seeps into homes. Exposures due to inhalation of radon by people living and working indoors vary dramatically depending on the local geology, building construction and household lifestyles; this mode of exposure accounts for about half of the average human exposure to natural sources. It is now recognized that a very large number of workers are exposed to natural sources in their working places.

Concerns on nuclear test explosions in the atmosphere were the original reason for the UNSCEAR conception. They had been conducted at a number of sites, mostly in the northern hemisphere (the most active testing being in the periods 1952–1958 and 1961–1962), and the radioactive fallout from those tests represents a source of continuing exposure even today, albeit at very low levels. The most dominant peaceful exposure is medical exposure. Irrespective of the level of health care in a country, the medical uses of radiation continue to increase as techniques develop and become more widely disseminated; about 3.6 billion radiological examinations are conducted worldwide every year. (In countries with high levels of health care, exposure from medical uses is on average now equal to about 80% of that from natural sources.)

By contrast, radiation doses due to the generation of electrical energy by NPPs are extremely small in spite of the fact that this type of generation has grown steadily since 1956. Moreover, the doses due to the production of energy in the nuclear reactor are in turn a small part of the doses due to the nuclear fuel cycle, which includes the mining and milling of uranium ore, fuel fabrication, storage or reprocessing of irradiated fuel, and storage and disposal of radioactive wastes. The doses to which the public are exposed vary widely from one type of fuel-cycle installation to another, but in any case they are generally small and they decrease the further the distance from the facility. Moreover, they have been markedly reduced over time because of lower discharge levels. For instance, Fig. 11.2 presents the reduction of normalized noble gas releases for different periods and types of reactor. Over the period 1970–2002, radioactive releases (expressed as 10^{12} becquerels per 10^9 watts of electrical energy produced) of noble gases were



11.2 Normalized noble gas releases for different periods and types of reactor.

reduced from 13,000 to 112; those for tritium were reduced from 448 to 43, and for iodine from 0.047 to 0.0006 (UNSCEAR, 2011).

In sum, the doses due to the nuclear fuel cycle in general and to NPPs in particular are a tiny fraction of the doses incurred by the population. Table 11.3 presents the latest UNSCEAR estimates of global radiation doses from different sources (UNSCEAR, 2011).

Occupational exposures

The workforce exposed to radiation in NPPs amounts to several hundred thousand workers. This is a fraction of the total of occupationally exposed workers in nuclear fuel cycle activities and a relatively small fraction of those exposed in other non-nuclear activities. The total number of occupationally exposed workers is estimated by UNSCEAR to be about 22.8 million, of whom about 13 million are exposed to enhanced natural sources of radiation, about 7.5 million to medical sources and the rest to other artificial sources. Table 11.4 presents some activities or practices involving occupational exposure (UNSCEAR, 2011).

The major occupational exposure for the production of nuclear electricity has not been occurring at NPPs but rather in other parts of the nuclear fuel

Table 11.3 Annual average doses and ranges of individual doses by source

Source or mode	Annual average dose (worldwide) (mSv[a])	Typical range of individual doses (mSv[a])	Comments
Natural sources of exposure			
Inhalation (radon gas)	1.26	0.2–10	The dose is much higher in some dwellings.
External terrestrial	0.48	0.3–1	The dose is higher in some locations.
Ingestion	0.29	0.2–1	
Cosmic radiation	0.39	0.3–1	The dose increases with altitude.
Total natural	2.4	1–13	**Sizeable population groups receive 10–20 mSv.**
Artificial sources of exposure			
Medical diagnosis (not therapy)	0.6	0-several tens	The averages for different levels of health care range from 0.03 to 2.0 mSv; averages for some countries are higher than that due to natural sources; individual doses depend on specific examinations.

Atmospheric nuclear testing	0.005	Some higher doses around test sites still occur	The average has fallen from a peak of 0.11 mSv in 1963.
Occupational exposure	0.005	~0–20	The average does to all workers is 0.7 mSv. Most of the average dose and most high exposures are due to natural radiation (specifically radon in mines).
Chernobyl accident	0.002[b]	In 1986, the average dose to more than 300,000 recovery workers was nearly 150 mSv; and more than 350,000 other individuals received doses greater than 10 mSv	The average in the northern hemisphere has decreased from a maximum of 0.04 mSv in 1986. Thyroid doses were much higher.
Nuclear fuel cycle (public exposure)	0.0002[b]	Doses are up to 0.02 mSv for critical groups at 1 km from some nuclear reactor sites	
Total artificial	0.6	From essentially zero to several tens	Individual doses depend primarily on medical treatment, occupational exposure and proximity to test or accident sites.

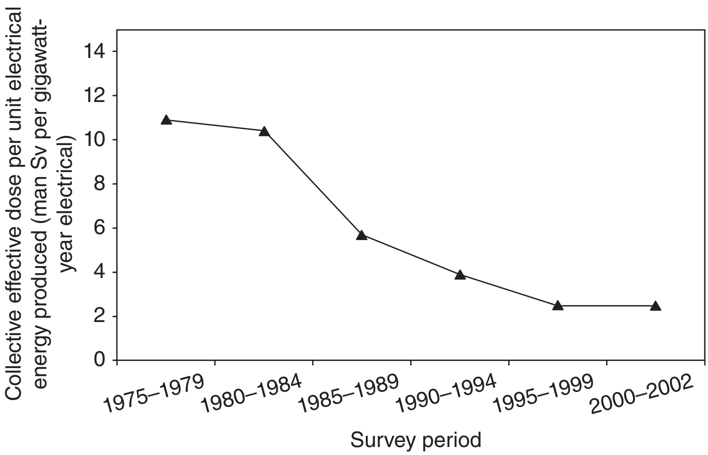
[a] In millisieverts (mSv), the unit of measurement of effective dose.
[b] Globally dispersed radionuclides. The value for the nuclear fuel cycle represents the maximum per capita annual dose to the public in the future, assuming the practice continues for 100 years, and derivers mainly from globally dispersed, long-lived radionuclides released during reprocessing of nuclear fuel and nuclear power plant operation.

Table 11.4 Practices involving occupational exposure

Category of practice	Practice
Exposure to natural sources of radiation	Civilian aviation
	Coal mining
	Other mineral mining
	Oil and natural gas industries
	Workplace exposure to radon other than in mines
Nuclear fuel cycle	Uranium mining
	Uranium enrichment and conversion
	Fuel fabrication
	Reactor operation
	Decommissioning
	Fuel reprocessing
	Research in the nuclear fuel cycle
	Waste management
Medical uses	Diagnostic radiology
	Dental radiology
	Nuclear medicine
	Radiotherapy
	All other medical uses
Industrial uses	Industrial irradiation
	Industrial radiography
	Luminizing
	Radioisotope production
	Well logging
	Accelerator operation
	All other industrial uses
Miscellaneous	Educational establishments
	Veterinary medicine
	Other occupations
Military activities	All military activities

cycle, mainly in the mining of uranium. The average annual effective dose incurred by workers involved in the nuclear fuel cycle has gradually declined since 1975, from 4.4 mSv to 1.0 mSv at present. Much of this decline is because of the more advanced mining techniques in uranium mining. Concurrently, the lower occupational exposures at NPPs have also been declining, mainly due to optimization of radiation protection (see hereinafter). UNSCEAR reports that in the period 1970–2002 the individual dose declined from a higher average of 13.2 millisievert per annum to a lower of 0.2 millisievert per annum. As shown in Fig. 11.3, the total occupational exposure at NPPs divided by the energy produced has also fallen steadily over the past three decades (UNSCEAR, 2011).

On the other hand, the trends in average annual occupational effective doses show a clear increase for the occupations being exposed to natural



11.3 Annual occupational collective dose at reactors.

Table 11.5 Trends in average annual occupational effective doses in millisieverts per annum (1980–1984, 1990–1994 and 2000–2002)

Source of exposure	1980–1984	1990–1994	2000–2002
Natural sources	–	1.8	2.9
Military activities	0.7	0.2	0.1
Nuclear fuel cycle	3.7	1.8	1.0
Medical uses	0.6	0.3	0.5
Industrial uses	1.4	0.5	0.3
Miscellaneous	0.3	0.1	0.1

radiation and a decrease for the nuclear fuel cycle. This is shown in Table 11.5.

11.3 Biological effects of radiation

In no other field of scientific investigation does an international mechanism to achieve global consensus exist compared with that specifically set up for estimating health effects attributable to exposure to ionizing radiation. UNSCEAR has, for nearly half a century, annually assembled leading radiation specialists to provide the most plausible estimates of the health risks attributable to radiation exposure, and periodically submitted them to the 192 world governments via the UN General Assembly. The extremely detailed UNSCEAR reports on radiation effects are a synthesis of thousands of peer-reviewed references. While it is certainly unfeasible to summarize accurately such a vast amount of information, the author has made several brief accounts of UNSCEAR estimates aimed at a broad audience (González, 2002, 2004b, 2004c).

UNSCEAR's estimates have not changed substantially over the past years and can be categorized into two types of effects, namely (1) prompt tissue-reactions that are usually termed '*deterministic*' effects, because they are determined to occur above certain dose thresholds, and (2) long-term late effects, such as cancer, which are termed '*stochastic*' effects due to the aleatory nature of their manifestation.

11.3.1 Tissue reactions: deterministic effects

At high levels of radiation doses, the cell-killing properties of radiation exposure will cause tissue-reaction effects that are usually termed 'deterministic' effects, because they are determined to occur above a certain dose. In fact, the induction of tissue reactions is generally characterized by a threshold dose. The reason for the presence of this threshold dose is that radiation damage (serious malfunction or death) of a critical population of cells in a given tissue needs to be sustained before injury is expressed in a clinically relevant form. Above the threshold dose the severity of the injury, including impairment of the capacity for tissue recovery, increases with dose. These effects can be clinically diagnosed in the exposed individual.

Table 11.6 presents the projected threshold estimates of the acute absorbed doses for 1% incidences of morbidity and mortality involving adult human organs and tissues after whole-body exposures to gamma rays similar to those encountered in NPPs. It is to be noted that these levels of acute absorbed doses can be reached in NPPs only if a serious accident occurs. These levels are inconceivable in normal operations.

11.3.2 Stochastic effects

On the other hand, it has been widely postulated that any radiation exposure, at any level, however small, may cause a risk for future increases in the natural incidence of some malignancies and hereditable effects. On the basis of available radio-epidemiological studies in humans exposed to relatively high radiation doses, UNSCEAR has assessed that the excess lifetime risk of mortality (averaged over both sexes) is:

- for all solid cancers combined, 3.6–7.7% per Sv for an acute dose of 0.1 Sv, and 4.3–7.2% per Sv for an acute dose of 1 Sv
- for leukaemia, 0.3–0.5% per Sv for an acute dose of 0.1 Sv, and 0.6–1.0% per Sv for an acute dose of 1 Sv.

Taking into account available radio-biological information and epidemiological studies in animals, UNSCEAR has also made estimates of risk of

Table 11.6 Projected threshold estimates of acute absorbed doses (for 1% incidences of morbidity and mortality involving adult human organs and tissues after whole-body gamma ray exposures)

Effect	Organ/tissue	Time to develop effect	Absorbed dose (Gy)[e]
Morbidity:			*1% incidence*
Temporary sterility	Testes	3–9 weeks	~0.1[a,b]
Permanent sterility	Testes	3 weeks	~6[a,b]
Permanent sterility	Ovaries	<1 week	~3[a,b]
Depression or blood-forming process	Bone marrow	3–7 days	~0.5[a,b]
Main phase or skin reddening	Skin (large areas)	1–4 weeks	<3–6[b]
Skin burns	Skin (large areas)	2–3 weeks	5–10[b]
Temporary hair loss	Skin	2–3 weeks	~4[b]
Cataract (visual impairment)	Eye	Several years	~1.5[a,c]
Mortality:			
Bone marrow syndrome:			
– without medical care	Bone marrow	30–60 days	~1[b]
– with good medical care	Bone marrow	30–60 days	2–3[b,d]
Gastro-intestinal syndrome:			
– without medical care	Small intestine	6–9 days	~6[d]
– with good medical care	Small intestine	6–9 days	>6[b,c,d]
Pneumonitis	Lung	1–7 months	6[b,c,d]

[a] ICRP (1984).
[b] UNSCEAR (1988).
[c] Edwards and Lloyd (1996).
[d] Scott and Hahn (1989), Scott (1993).
[e] Most values rounded to the nearest gray; ranges indicate area dependence for skin and differing medical support for bone marrow.

heritable diseases in one generation due to low-dose exposure and concluded that the risks in the first generation (per unit low-LET dose) are:

- for dominant effects (including X-linked diseases), ~750–1500 per million per gray *vis-à-vis* a baseline frequency of 16,500 per million
- for chronic multifactorial diseases, ~250–1200 per million per gray *vis-à-vis* a baseline frequency of 650,000 per million
- for congenital abnormalities, ~2000 per million per gray *vis-à-vis* a baseline frequency of 60,000 per million (chromosomal effects were assumed to be subsumed in part under the risk of autosomal dominant and X-linked diseases and in part under that of congenital abnormalities).

In sum, as far as radiation-induced heritable diseases is concerned, UNSCEAR concluded that for a population exposed to radiation in one generation only, the risks to the progeny of the first post-radiation generation are estimated to be 3000 to 4700 cases per gray per one million progeny, which constitutes 0.4–0.6% of the baseline frequency of those disorders in the human population.

It should be noted, however, that these estimates are associated with unavoidable uncertainties. The processes occurring from the ionization of living matter by radiation exposure up to the expression of the attributable detrimental health effects are extremely complicated and can only be assessed with considerable uncertainties. For stochastic effects they extend over different time periods: the physical interaction taking place in millionths of microseconds, the physicochemical interactions occurring in thousandths of microseconds up to milliseconds, the biological response arising in seconds up to days, and the stochastic medical effects expressed after years, decades and – in the case of hereditary effects – probably centuries.

11.3.3 Summary of the biological effects of radiation

In summary,

- In normal operations of NPPs radiation doses incurred by the members of the public will be insignificant and those incurred by workers will be relatively small, lower than the typically elevated levels of the background radiation that is ubiquitous in nature.
- These radiation doses are far below the threshold doses of deterministic effects. Therefore, the occurrence of deterministic effects in NPPs is prevented.
- Nonetheless, it is assumed that low radiation doses have the potential to induce stochastic effects, such as cancer and hereditable harm, that may become manifest many years after the exposure; the probability of occurrence of stochastic effects at low doses is exceedingly small, although it is assumed to increase proportionally with dose, and the effects are unlikely to be detectable (ICRP, 2005b; Beninson, 1996).
- Conversely, workers involved in an accident within an NPP (perhaps only a small number of the workforce) could also be exposed to high radiation doses, e.g. of the order of thousands of millisieverts. If such dose levels are incurred, clinically visible deterministic health effects are almost certain to appear, usually as burns and other tissue reactions, within days of the exposure, affecting the functioning of tissues and organs with a severity that increases with dose. In severe cases, they can cause the death of exposed individuals.

The effects of different radiation doses and the likelihood of observable consequences are summarized in an extremely simplified manner in Table 11.7 (ICRP, 2005a).

Table 11.7 Summary of radiation-induced health effects

Dose	Effects on individuals	Consequences for an exposed population
Very low dose: about 10 mSv (effective dose) or less	No acute effects; extremely small additional cancer risk	No observable increase in the incidence of cancer, even in a large exposed group
Low dose: towards 100 mSv (effective dose)	No acute effects, subsequent additional cancer risk of less than 1%	Possible observable increase in the incidence of cancer, if the exposed group is large (perhaps greater than about 100,000 people)
Moderate dose: towards 1000 mSv (acute whole-body dose)	Nausea, vomiting possible, mild bone marrow depression; subsequent additional cancer risk of about 10%	Probable observable increase in the incidence of cancer, if the exposed group is more than a few hundred people
High dose: above 1000 mSv (acute whole-body dose)	Certain nausea, likely bone marrow syndrome; high risk of death from about 4000 mSv of acute whole-body dose without medical treatment. Significant additional cancer risk	Observable increase in the incidence of cancer

11.3.4 Nominal risk coefficients

Taking into account the above described UNSCEAR estimates for the effects that can be attributed to the normal operation of NPPs, and its own findings, the ICRP recommended the use of '*detriment-adjusted nominal risk coefficients*' for the only purpose of radiological protection at low doses. These coefficients are numerals expressed in % per unit dose, which – multiplied by dose – aim at quantifying the plausibility or 'degree of belief' of latent effects as a result of radiation exposure. The coefficients are nominal, in the sense that they do not necessarily correspond to a real value, since they relate to hypothetical (not real) people who are averaged over age and sex. Since the different possible effects may cause distinct detriment to people, the coefficients are multidimensional, quantifying the plausible expectation of harm, and including among other factors the weighted plausibility of fatal and non-fatal harm, and life lost should the harm actually occur.

The detriment-adjusted nominal risk coefficients recommended by ICRP are:

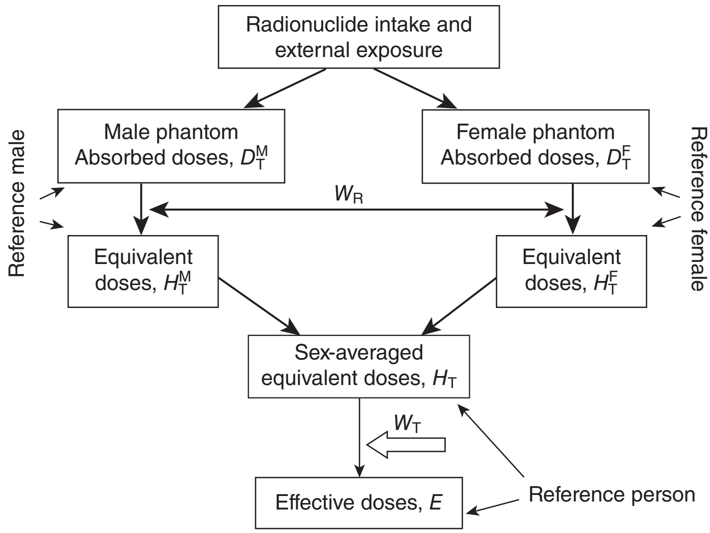
- for malignancies,
  - 5.5% Sv^{-1} for a whole population
  - 4.1% Sv^{-1} for an adult population
- for hereditable effects,
  - 0.2% Sv^{-1} for a whole population
  - 0.1% Sv^{-1} for an adult population

which result in a combined value of

- 5.7% Sv^{-1} for a whole population
- 4.2% Sv^{-1} for an adult population.

The risk coefficients imply a central assumption of a linear dose–response relationship for the induction of cancer and heritable effects, according to which an increment in dose would induce a proportional increment in risk even at low doses. This assumption is essential for the practical implementation of the system of radiation protection (see hereinafter) in order to provide the basis for the summation of doses of various levels, from different sources, and from external exposure and from intakes of radionuclides (Beninson, 1996).

It is again emphasized that the risk coefficients are *nominal*, i.e. artificially constructed using average phantoms. Figure 11.4 indicates how the effective dose is constructed using these phantoms (ICRP, 2007b).



11.4 Use of phantoms and sex averaging to obtain the effective dose.

International radiation safety standards have taken the UNSCEAR estimates and ICRP recommendations into account, rounding an overall nominal risk coefficients to ~5% Sv^{-1}, as the basis of the requirements for limiting radiation risks. This is because, while it is not demonstrable, it is considered plausible that risks be attributable to radiation exposures, even at low doses, and therefore for reasons of social duty, responsibility, utility, prudence and precaution, it is ethically required that regulatory bodies do ascribe such nominal radiation risks to prospective exposure situations. However, both UNSCEAR and ICRP had made clear that while nominal risk coefficients can be used for attributing risk for purposes of prospective planning, they cannot be used for attributing factual health effects retrospectively.

11.4 Attributability of risks and potential health effects to nuclear power plants (NPPs)

Attribution of radiation risks and effects to radiation exposure situations, particularly those involving the low doses registered at NPPs, is a tricky issue. It means regarding actual effects or postulated risk, or both, as being caused by the exposure situation and assigning them to the situation, therefore transferring to the situation the related responsibilities and liabilities (the term is rooted in the Latin *attribuere*, from *ad*- 'to' + *tribuere* 'assign').

However, this chapter is not intended to address issues of law involving imputation rather than attribution; e.g., it does not deal with the legal concept of causality that is common in occupational litigations (ILO, 2010). Rather, its aims are to focus on the epistemology of the issue, namely on the current theories of knowledge on health effects of radiation at low doses, especially with regard to the methods, validity and scope of the theories. From this epistemological basis, it endeavours to clarify a conundrum in radiation sciences: whether radiation risks or radiation effects, or both, are attributable to NPP operations.

While the attribution of radiation risk is associated with the concept of *probability*, the attribution of radiation effects should be based on the concept of *provability*. These two concepts are subtly distinct: probability is an established quantity, measurable through statistical techniques or assignable through formal Bayesian approaches, which measure the likelihood that harm might be incurred; conversely, provability appears to be an unquantifiable quality describing the capability to demonstrate by evidence the actual occurrence of radiation effects. Thus, on the basis of the available evidence, attribution notions should be elucidated. These should be apparent to experts but are usually not substantiated on epistemology and seemingly remain obscure for decision-makers and the general public.

11.4.1 Attributability of deterministic effects

Deterministic effects can be attributed to specific NPP exposures with a high degree of confidence under the following conditions:

- The dose incurred was higher than the relevant dose-threshold for the specific effect.
- In addition, an unequivocal pathological diagnosis is attainable ensuring that possible competing causes have been eliminated.

Only under both of these conditions may the occurrence of the effect be properly attested and attributed to the exposure. One exception to this general rule could be specific situations of radiation exposure to the lens of the eye that might be sufficient to induce opacities, a situation that may be familiar in interventional radiology but should not occur at NPPs.

11.4.2 Attributability of stochastic effects

Conversely, malignant or hereditary effects cannot be unequivocally attributed to radiation exposure on individual bases for reasons of counterfactual conditionality. This is because radiation exposure is not the only possible cause of these types of effects and, at present, no biomarkers are available for these effects that are specific to radiation exposure.

However, while the occurrence of malignant effects (or of hereditable effects in the descendants of those exposed) cannot be unequivocally attributable to radiation on an individual basis, an increased incidence of these effects in a population can theoretically be attributed to radiation on a collective basis. Collective attribution can be established through epidemiological analysis, under the following conditions:

- The number of cases of the effect in the exposed population should be sufficient to overcome the inherent aleatory uncertainties of epidemiological analyses.
- In addition, the increase in the collective prevalence of the effect in the exposed population is properly attested by a qualified radio-epidemiological procedure.

In situations of chronic exposures at levels similar to those arising from normal operations of NPPs, the expected number of additional cases of malignancies in a commensurate population for an epidemiological study would be so low that attribution is unattainable either individually or collectively.

Thus, while increased incidences of malignancies and hereditary effects might theoretically occur in populations exposed to NPPs, since it is not feasible to obtain unequivocal scientific evidence of their occurrence, they therefore should neither be deemed attributable nor be used prospectively in notional projections of radiation harm. Moreover, hereditary effects cannot at present be attributed to radiation exposure, even at high doses, because the fluctuation in the normal incidence of these effects is likely to be so much larger than any expected radiation-related increase in the incidence.

It is to be noted, however, that occasionally individual attribution can nevertheless be *ostensible*, namely, apparently factual, even if not necessarily so; this may be the case when:

- the 'background' incidence of the effect is low, and
- the radio-sensitivity of the effect is high.

A typical example of ostensible individual attributibility is the case of follicular thyroid cancer in children exposed to relatively high thyroid doses such as those incurred after the Chernobyl accident.

11.4.3 Attributability of cellular damage

It should moreover be noted that effects can occur in human cells exposed to relatively high levels of radiation. These effects can be detected through specialized bioassay specimens, such as some haematological and cytogenic sampling. They may be used as biological indicators of the exposure and can help to identify and even quantify high individual exposures to radiation, such as those occurring in accidents. However, the presence of biological indicators of exposure does not necessarily imply that the individual had experienced or would experience health effects that could be attributed to radiation.

11.4.4 Attributability of radiation risk

Notwithstanding the unattributability of stochastic effects, it should be noted that under present knowledge, it can be demonstrated that *risks* (rather than factual effects) can in fact be attributable to radiation exposure situations, even to those delivering small doses. Therefore, for reasons of radiation protection, and also of duty, responsibility, prudence and precaution, it is necessary to ascribe nominal radiation risks to prospective exposure situations. Thus, nominal risk coefficients should be developed from the observed increased incidence of radiation effects at high doses and be used solely for radiation protection purposes (see hereinafter).

11.5 Radiation protection paradigm

The radiation protection paradigm is a model for keeping people safe from radiation injury or harm, which in this case could be caused by NPP operations. It is founded on fundamental principles, which in turn are based on solid ethical doctrines, and built up into a system of radiation protection. The primary aim is to achieve an appropriate level of protection for people and the environment against the detrimental effects of radiation exposure without unduly limiting the desirable human actions that may be associated with such exposure, one of these actions being the generation of nuclear electricity. The system includes a classification of feasible exposure situations, a characterization of type of exposures and a scheme for controlling such exposures.

It is to be noted that radiation protection concerns all exposures to radiation from any source, regardless of its size and origin. However, the restraint of exposures can apply in their entirety only to situations in which either the source of exposure or the pathways leading to the doses received by individuals can be controlled by some reasonable means. Some exposure situations are excluded from radiological protection legislation, usually on the basis that they are unamenable to control with regulatory instruments (e.g., some exposure to natural sources), but this is not the case for exposure situations from NPPs which are unexceptionally included in regulations. However, some exposure situations at NPPs may be exempted from some radiation protection regulatory requirements whenever such controls are regarded as unwarranted, e.g. because the activity of the sources and the exposure they deliver are minute. ICRP has issued comprehensive recommendations for exclusion and exemptions (ICRP, 2007b). Remarkably, international agreements have been reached for exemption values in all commodities (IAEA, 2004b, 2004c), for drinking water (WHO, 2004) and for foodstuffs (CAC, 2006).

11.5.1 The basic principles

The basic principles developed by ICRP over the years continue to be regarded as the fundamental basis for a system of radiological protection (ICRP, 2007a). They can be simplistically formulated as follows:

- *Principle of justification* (of actions that modify the radiation exposure people incur): Any decision that alters the radiation exposure situation should do more good than harm – meaning that by introducing new radiation sources or by intervening for reducing existing doses (or the risk of potential doses), sufficient individual or societal benefit should be achieved to offset the radiation detriment such actions may cause. In the case of NPPs this principle could be translated as follows: the introduction of a NPP should provide sufficient benefits to the society and its individuals as to offset the radiation detriment that the NPP operation might cause. (The need for justification of NPPs has been treated in Chapter 8.)

- *Principle of optimization* (of radiation protection): The likelihood of incurring exposures, the number of people exposed, and the magnitude of their individual doses and risks should all be kept as low as reasonably achievable, taking into account economic and societal factors. For NPPs this can be formulated as follows: the level of radiation protection designed for the NPP and the level of radiation protection during its operation should be the best under the prevailing circumstances, maximizing the margin of benefit over harm.
- *Principle of individual protection*: Inequitable individual protection outcomes of justification and optimization should be prevented by restricting individual doses, by applying individual-related *dose limits* and source-related *dose constraints* and *reference levels*. For NPPs, plant-related dose constraints and reference levels should be established respecting individual-related dose limits.

These principles contain embedded values of prudence encompassing the protection of future generations and their habitat. These values can be formulated as a *de facto* principle:

- *Principle of intergenerational prudence*, which extends the radiological protection principles to all humanity, regardless of where and when they live, and implies that all humans, present and future, and their habitat shall be afforded a level of protection that is not weaker than the level provided to the populations of the society causing the protection needs. In practice, this means that the dose from NPPs to be controlled is the committed dose rather than the incurred dose.

11.5.2 The ethical basis of the radiation protection principles

There is a direct correlation between the basic principles of radiological protection recommended by ICRP and basic universal ethical doctrines. This correlation can be described as follows (González, 2010):

- The principle of *justification* is based on *teleological ethics* (namely consequentiality ethics), which is expressed with the aphorism 'Mind the ends, which justify the means'.
- The principle of *optimization* is based on *utilitarian ethics*, which is expressed with the aphorism 'Do the greatest good for the largest number of people'.
- The principle of *limitation* is based on *deontological ethics*, which is expressed with the aphorism 'Do not unto others what they should not do unto you' (or with the religious dogma 'God's commandments may be summed up in this one rule: care for your neighbour as yourself').

- The principle of *intergenerational prudence* is based on *aretaic ethics* (namely virtue ethics) which is expressed with the aphorism 'Do good that will not be returned' and is the basis for complying with the UN's Precautionary Principle (UNESCO, 2005).

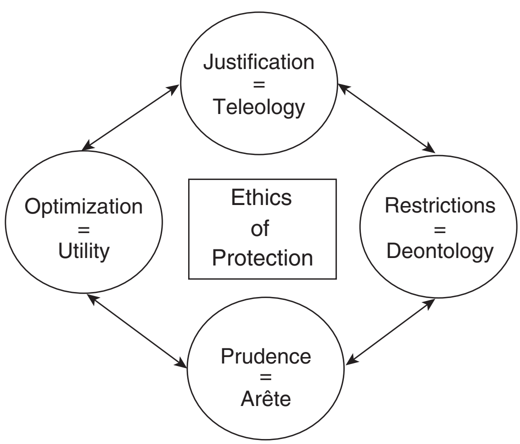
Teleological and utilitarian ethics belong to a family of 'social-oriented' ethics; deontological and aretaic ethics belong to a family of 'individual-oriented' ethics. In relation to radiation protection, namely for keeping humans safe from radiation harm or injury, teleological and utilitarian ethics would aim at the principles for protecting society as a whole, while deontological and virtue ethics are more focused on individual protection.

The principles and their ethical foundations are interrelated and applicable to *all* exposure to radiation risk, namely to exposures to 'certain' doses and to exposures to the 'potential' of doses. Moreover, distinctly from conventional ethical approaches, the ICRP principles harmonize all the prevailing ethical doctrines and use *all* of them in conjunction, as illustrated in Fig. 11.5 (González, 2010).

Building bridges among the ethical doctrines and applying them to radiological protection has historically been at the roots of the ethic accomplishment of the ICRP recommendations.

11.5.3 Classification of radiation exposure situations

Radiation protection applies to all conceivable radiation exposure situations, which can be classified as planned, emergency and existing exposure situations, as follows (ICRP, 2007a):



11.5 The radiation protection principles and their ethical foundations.

- *Planned exposure situations* refer to circumstances involving the planned introduction and operation of sources that may expose people to radiation.
- *Emergency exposure situations* refer to unexpected accidental conditions that may occur during the operation of a planned situation, or from a malicious act, and which require urgent protective attention.
- *Existing exposure situations* refer to a radiation environment that already exists when a decision on control has to be taken, e.g., natural background radiation exposure situations.

For NPPs, the design and operation stages are clearly planned exposure situations. If an accident occurs, it should be treated as an emergency exposure situation. The residual risks that might remain after the decommissioning and closure of NPPs may be treated as an existing exposure situation (González, 2009b).

11.5.4 Categorization of individual exposures

Three main categories of exposures can be distinguished (ICRP, 2007a):

1. *Occupational exposures*, which are all exposures incurred by workers in the course of their work, with the exception of
   - excluded exposures and exposures from exempt activities involving radiation or exempt sources
   - any medical exposures
   - the normal local natural background radiation.
2. *Public exposures*, which are exposures incurred by members of the public from radiation sources, excluding any occupational or medical exposure and the normal local natural background radiation.
3. *Medical exposures* of patients.

Exposures of comforters and carers, and exposures of volunteers in research, are treated separately in radiation protection. From the above categories, the only ones relevant for NPPs are occupational and public exposures.

It is to be noted that while the categorization of exposure does not recognize gender distinctions, if a female worker at the NPP has declared that she is pregnant or nursing, additional controls have to be considered in order to attain a level of protection for the embryo/fetus broadly similar to that provided for members of the public.

11.5.5 Control of exposures

Exposure levels should be controlled through *dose limits*, *dose constraints* and *reference levels* for representative individuals (ICRP, 2007a) and mainly throughout the full application of the principle of optimization of protection. ICRP has provided ample guidance for the implementation of optimization (ICRP, 1973, 1980, 1990, 2006b).

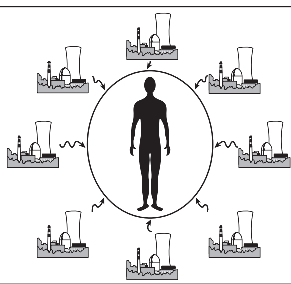
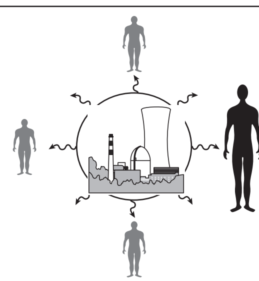
A *dose limit* is an individual-related dose restriction defined as the value of the effective dose or the equivalent dose to individuals from planned exposure situations that shall not be exceeded.

Dose constraints are prospective and source-related restrictions on the individual dose from a given source, which provide a basic level of protection for the most highly exposed individuals from that source, e.g. a NPP *in toto* or any of its systems, and serve as an upper bound on the dose in optimization of protection for that source. For occupational exposures, the dose constraint is a value of the individual dose used to limit the range of options considered in the process of optimization. For public exposure, the dose constraint is an upper bound on the annual doses that members of the public should receive from the planned operation of the NPP.

Reference levels are used in emergency or existing controllable exposure situations, and represent the level of dose or risk above which it is judged to be inappropriate to plan to allow exposures to occur, and below which optimization of protection should be implemented. The chosen value for a reference level will depend upon the prevailing circumstances of the exposure under consideration. They shall be used in the emergency planning of any NPP.

Figure 11.6 contrasts dose limits with dose constraints and reference levels for protecting workers and members of the public (ICRP, 2007a).

Table 11.8 illustrates the use of dose limits, dose constraints and reference levels within the system of protection for occupational and public exposures

Dose limits	Constraints and reference levels
Protect individual workers from occupational exposure and the representative person from public exposure	
	
From all regulated sources in planned exposure situations	From a source in all exposure situations

11.6 Dose limits *vis-à-vis* dose constraints and reference levels.

Table 11.8 Use of dose limits, dose constraints and reference levels

Type of situation	Occupational exposure	Public exposure
Planned exposure	Dose limit Dose constraint	Dose limit Dose constraint
Emergency exposure	Reference level[a]	Reference level

[a] Long-term recovery operations should be treated as part of planned occupational exposure.

at NPPs (ICRP, 2007a). The recommended dose limits are as follows (ICRP, 2007a):

- For occupational exposure in planned exposure situations, the limit should be expressed as an effective dose of 20 mSv per year, averaged over defined five-year periods (100 mSv in five years), with the further provision that the effective dose should not exceed 50 mSv in any single year.
- For public exposure in planned exposure situations, the limit should be expressed as an effective dose of 1 mSv in a year. However, in special circumstances a higher value of effective dose could be allowed in a single year, provided that the average over defined five-year periods does not exceed 1 mSv per year.

The limits on effective dose apply to the sum of doses due to external exposures and committed doses from internal exposures due to intakes of radionuclides. Occupational intakes may be averaged over a period of five years to provide some flexibility. Similarly, averaging of public intakes over a period of five years would be acceptable in such special circumstances where averaging of the dose to members of the public could be allowed.

Finally, Table 11.9 presents the framework for recommended source-related dose constraints and reference levels with examples of constraints for workers and the public exposed to NPPs.

11.5.6 Protection of the environment

Current radiation protection approaches acknowledge the importance of protecting not only humans but also the environment. Previously the concern focused on mankind's environment only with regard to the transfer of radionuclides through it, mainly in the context of planned exposure situations. In such situations, the standards of environmental control needed to protect the general public would ensure that other species are not placed at risk. To provide a sound framework for environmental protection in all exposure situations, there has been proposed the use of 'reference animals

Table 11.9 Framework for recommended source-related dose constraints and reference levels

Bands or constraints and reference levels[a] (mSv)	Characteristics of the exposure situation	Radiological protection requirements	Examples
Greater than 20 to 100[b,c]	Individuals are exposed by sources that are not controllable, or where actions to reduce doses would be disproportionately disruptive. Exposures are usually controlled by action on the exposure pathways.	Consideration should be given to reducing doses. Increasing efforts should be made to reduce doses as they approach 100 mSv. Individuals should receive information on radiation risk and on the actions to reduce doses. Assessment of individual doses should be undertaken.	Reference level set for the highest planned residual dose from a radiological emergency.
Greater than 1 to 20	Individuals will usually receive benefit from the exposure situation but not necessarily from the exposure itself. Exposures may be controlled at source or, alternatively, by action in the exposure pathways.	Where possible, general information should be made available to enable individuals to reduce their doses. For planned situations, individual assessment of exposure and training should take place.	Constraints set for occupational exposure in planned situations.
1 or less	Individuals arc exposed to a source that gives them little or no individual benefit but benefits to society in general. Exposures are usually controlled by action taken directly on the source for which radiological protection requirements can be planned in advance.	General information on the level of exposure should be made available. Periodic checks should be made on the exposure pathways as to the level of exposure.	Constraints set for public exposure in planned situations.

[a] Acute or annual dose.
[b] In exceptional situations, informed volunteer workers may receive doses above this band to save lives, prevent severe radiation-induced health effects, or prevent the development of catastrophic conditions.
[c] Situations in which the dose threshold for deterministic effects in relevant organs or tissues could be exceeded should always require action.

and plants'. In order to establish a basis for acceptability, additional doses calculated to these reference organisms could be compared with doses known to have specific biological effects and with dose rates normally experienced in the natural environment. Nobody, however, is proposing to set any form of 'dose limits' for environmental protection.

It should be recognized that until recently the word *environment* itself was absent in normal parlance and, unsurprisingly, concerns for environmental protection are a relatively new phenomenon. The term 'environment' derives from the old French *environ*, 'surroundings', from *en* 'in' + *viron* 'circuit', strictly referring to the surroundings of an object. More recently it has evolved to mean the surroundings or conditions in which a person, animal or plant lives or operates and, even more recently, it has become equated to the natural world, especially as affected by human activity. It will certainly take time to develop comprehensive protection doctrines for such a relatively contemporary concept, one that encompasses this relatively new human apprehension. Over the last years, two fundamental environmental protection approaches (rather than ethics) are being constructed: the so-termed biocentrism and ecocentrism.

In spite of this apparent vacuum of an environmental protection ethics, some basic principles are being developed for protecting not only humans but also the environment in itself from the detrimental effects of radiation exposure. The aim is to ensure that the development and application of approaches to environmental protection are compatible with those for radiological protection of humans, and with those for protection of the environment from other potential hazards (IAEA, 2005b).

As indicated heretofore, within the context of planned exposure situations, the standards of environmental control needed to protect the general public should ensure that other species in the human habitat are not placed at risk. However, the situation could be different in emergency and existing situations and in the environment at large. Thus, the radiation protection community is adhering to some international basic environmental protection objectives such as:

- to maintain biological diversity
- to ensure the conservation of species
- to protect the health and status of natural habitats, communities and ecosystems.

Under these premises, a framework for assessing the impact of ionizing radiation on non-human species (ICRP, 2003) and the techniques for implementation (ICRP, 2008) have been recommended by ICRP.

Ultimately, the protection of the environment from radiation exposure will be achieved through international efforts for restricting discharges of radioactive substances (González, 2005).

11.6 Potential exposures

While the radiation protection principles were originally formulated for dealing with protection against '*certain*' exposures, namely against exposures that will occur with some degree of certainty, they may, *mutatis mutandi*, be used against '*potential*' exposures as well, namely against situations having the capacity to develop into real exposures in the future. Namely, the principles described heretofore can be used not only for 'radiation protection' but also for 'radiation safety' in general and for nuclear safety in particular. Nuclear safety has been treated in Chapter 10.

Proposals for safety criteria for NPPs founded on the underlying radiation protection principles were suggested very early (González, 1974, 1982, 1986). The basic proposal was to use available probabilistic assessment tools, such as event and fault trees, for *a priori* overall safety analyses. A comparison could, therefore, be performed between the probability of occurrence of a hypothetical chain of events leading to an unexpected human exposure, along with its consequences in terms of doses incurred, and a regulatory criterion based on the radiation protection principles. The relevant regulatory authorities would then be able to judge safety levels on the basis of a rational approach sharing the same principles of radiological protection.

A conceptual framework for the protection from potential exposure and how to apply the conceptual framework to selected radiation sources has been recommended internationally (ICRP, 1993, 1997b).

There is at least one practical regulatory application of the radiation protection principles to a nuclear safety criterion (CNEA, 1979, 1980; ARN, 2010), which was discussed at various international meetings (González, 1982, 1986). The aim of the regulatory criterion is to require applicants for a NPP licence to identify the failure sequences which, in the case of occurrence, will deliver a radiation dose to members of the public, and make their probability of occurrence sufficiently low to be coherent and consistent with the radiological protection principles. The probability of occurrence of each failure sequence, as well as the corresponding activity of released radionuclides, should be assessed by using event and fault tree analyses, which must comply with the following criteria:

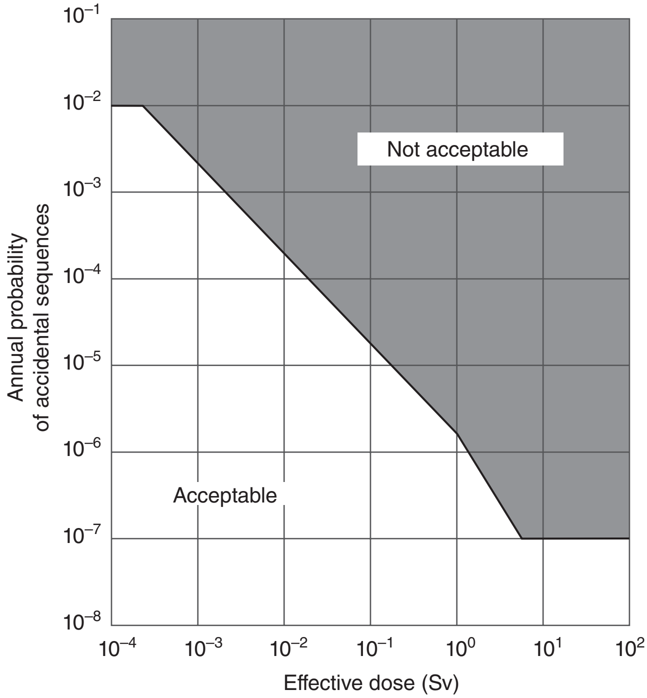
1. The failure analysis shall systematically encompass all foreseeable failures and failure sequences, including the common-mode failures, the failure combinations and the situations exceeding the design basis (failure in this context means an aleatory event preventing a component from performing its safety function, as well as any other event which may additionally occur as a necessary consequence of such deficiency; failure sequence, on the other hand, means a sequential series of failures which can, although not necessarily, occur after an initiating event.

2. A failure or a failure sequence may be selected as representative of a group of failures or of failure sequences (in such a case, the failure or failure sequence to be selected from the group shall be that delivering the worst consequences and the analysis shall take into account the sum of the probabilities of the failure or failure sequences in the group).
3. The analysis shall consider that a protection function may have lost operativeness either before the occurrence of the failure or of the failure sequence or as a result of such occurrence.
4. The analyses of failures, of failure sequences or of any part thereof shall be based on experimental data as far as possible (if this cannot be done, the valuation methods must be validated through appropriate tests).
5. The levels of failure rate assigned to the safe-related components, in the evaluation of the failure probability of systems, shall be justified. In the case that justifiable values were not available for some of the components, the applicant shall use levels of failure rate prescribed by the licensing authority (if a given failure rate is justified on the basis of quality assurance, this must be specified in detail).
6. The failure analyses shall consider the maintenance and testing procedures, and the time interval between successive maintenance and testing actions.
7. Failure rates postulated for human actions shall be justified taking into account the complexity of the task, the stress involved and any other factors which might influence that failure rate.

Thus annual probability of occurrence of any failure sequence, when plotted as a function of the resulting effective dose, shall result in compliance with a criterion that is coherent and consistent with the principles of radiological protection enunciated above. The implicit basic safety goal is a risk limit derived from the dose limitation system used for radiation protection purposes, which – as seen before – includes four principles: two of them are source-related (e.g. justification and optimization) and the other two are individual-related (e.g. individual limitation and intergenerational protection). These latter principles entail that the risk committed by individual sources should be low enough as to be automatically disregarded. The currently recommended dose limit of 1 mSv per year implies an annual risk limit of around 10^{-5} for any individual, even for the highest exposed one, as a result of performing all practices involving radiation exposure. However, since the dose limits relate to individuals, appropriate constraints for individual doses should be selected for each source of exposure. The dose constraint must be sufficiently lower than the relevant dose limit, so as to prevent individual exposure due to several sources from exceeding such limit. Therefore, the *de facto* annual limit of individual risk would become lower that the limit of around 10^{-5}. On the basis of the above limit and taking into account the uncertainties usually involved in probabilistic safety assessments, an annual risk limit for accidental exposures from nuclear installations should not exceed an order of 10^{-6}. This would be consistent with the principles involved in the currently enforced system of dose limitation. Moreover, accidental exposures may arise from a theoretically infinite number of accidental sequences, each one having a given probability of occurrence and delivering a given expected dose to the most exposed individual. The actual risk incurred by this individual will then result from the integration of the tail distribution of doses (i.e., the complement of the probability function of doses) times the probability of death provided the dose is incurred. The safety constraint should therefore be that the value of this integral be lower than 10^{-6} per annum.

The assessment of all possible accidental sequences involving radiation exposure is extremely difficult and practically impossible. Therefore, the regulator may be satisfied if around a tenth of the most relevant sequences are identified, assigning them an annual risk limit of 10^{-7}. Since each sequence may result in different doses, a criterion curve may be adopted, which is a relationship between the annual probability of sequence occurrence and the expected individual dose, each point of the curve representing a constant level of risk. This criterion curve is shown in Fig. 11.7 (Failure of a point to be under the criterion curve does not necessarily mean that the risk constraint is not met, because even in this case, the integral of the tail distribution could be lower than 10^{-6} annum.)

The logic behind the criterion curve is as follows. For the range of doses from which only stochastic effects of radiation can be incurred, the criterion curve must show a constant, negative, 45° slope in a log annual probability versus log individual dose coordinate axis plane. This would ensure that the annual probability of incurring the dose times the probability of death provided the dose is incurred (the latter being in the order of 10^{-2} per Sv) will be kept constant. One of the coordinate points in this part of the curve would obviously be {annual probability = $\sim 10^{-7}$ annum^{-1}; individual dose = 1 Sv}, because the product 10^{-5} annum^{-1} $\times$ 1 Sv $\times$ 10^{-2} Sv^{-1} results in an annual risk of 10^{-7} annum^{-1}. In the dose range where non-stochastic effects of radiation may occur (i.e., for individual doses higher than around 1 Sv), the slope of the curve should increase in order to take account of the higher risks of death at these levels of dose. For doses higher than approximately 6 Sv, the probability of death approaches unity. From this level to higher doses, the criterion curve should remain constant at an annual probability of 10^{-7} (because the exposed individual would inevitably die regardless of the level of the dose). Between the coordinate points defined by {annual probability = 10^{-5} annum^{-1}; individual dose = 1 Sv} and {annual probability = 10^{-7} annum^{-1}; individual dose = 6 Sv}, the criterion curve should show a shape similar to that of the relationship between the individual dose and



11.7 Criterion curve for prospective probabilistic safety assessments.

the frequency of death (which, at that range, is approximately S-shaped but, for the sake of simplification, the Authority has decided to approximate these two points by means of a linear-shaped relationship. Finally, the criterion curve has been truncated at an annual probability level of 10^{-2}, because the occurrence of incidents having a higher annual probability (regardless of the dose) should reasonably be expected to be unacceptable for any regulator.

It should be emphasized that the criterion curve is individual-related; i.e., it is intended to limit the risk-rate on the individual incurring the highest risk, but does not take into account the overall expected impact from accidental situations. The criterion assures a level of safety which is sufficient to ensure that an individual risk constraint, compatible with the philosophy of the dose limitation system, will not be exceeded. It fails, however, to answer positively the old question of the safety engineers, i.e. is such a safety level safe enough as to preclude further safety improvements? An installation complying with the criterion would equally consider whether it is imposing risks (lower than the 'acceptable' one) to few individuals, or whether many individuals would incur such risks. If an accident does occur, however, the overall radiological impact will be very different in each case, suggesting that the overall safety level might be lower in the second case than in the first one. Optimization may require further safety improvements in the second case. But, is this really necessary, providing the individual-related criterion is met? And, if so, on what basis can optimization be implemented? These questions are not simple to answer but a logical response would allow for complementing the probabilistic criterion based on individual risk considerations alone.

Radiation protection assessments use the concept of *radiation detriment*, namely the mathematical expectation of harm, to quantify the impact from a source of radiation exposure. The detriment is an extensive quantity that estimates the combined impact of deleterious effects resulting from exposure to a given radiation source. It is defined as the expectation of the harm to be incurred, taking into account the expected frequency and severity of each type of deleterious effect. The detriment incurred by one individual receiving a dose in the range of stochastic effects is proportional to the effective dose incurred, the proportionality factor being the probability that the individual will incur a deleterious effect as a result of the exposure. Therefore, in cases of actual exposures to low levels of dose, the total detriment is proportional to the sum of all the individual effective doses incurred, i.e., to the collective dose commitment (this latter quantity results from the time integration of the collective dose rate, which, in turn, results from the integral of the population spectrum in terms of effective dose rate incurred). It was therefore tempting to use a similar concept for measuring the expected impact from accidental exposures (Beninson and González, 1981). For potential accidental exposures, the concept of detriment may keep its theoretical meaning, although it would become a quantity of a second order of stochasticity. In such case, the probability of a given exposure, i.e., the combined probabilities of both an accidental release and an environmental condition (dispersion, deposition), should be introduced in the formulation and integrated over all possibilities. Then, if low doses were expected, the detriment should be proportional to the resulting mathematical expectation of the collective dose commitment. For higher doses, another component of the detriment should be added in order to take into account the non-stochastic effects of radiation.

This idea of using the detriment of a second order of stochasticity, and the related mathematical expectation of collective dose commitment, for quantifying the impact from accidental exposures is really appealing, as the concept would allow for optimizing safety, increasing it to a sufficiently high level that further improvement would not be worthwhile taking into account both the benefits achieved in terms of expected collective dose commitment reduction and the cost of obtaining such reduction. However, unfortunately, it was demonstrated (Beninson and Lindell, 1981) that, at very low probabilities, the detriment will lose its usefulness as a basis for decision-making.

In fact, in such cases the standard deviation of the result may be orders of magnitude higher than the actual expectation and the coefficient of variability would become very large. The detriment is then no longer a central measure of the distribution of harm and, in addition, the uncertainty of the detriment becomes too large to make it meaningful, even if the probability as such could be estimated by safety assessments with an accurate degree of certainty. At very low failure probabilities, the inherent uncertainty of the product of probability and consequences makes the use of this quantity rather doubtful. For these reasons, for potential accidental exposures the principles of justification and optimization are implemented in a less quantitative manner. The value assigned to the variables follows a utility function of probability and consequence. The utility function usually gives more weight to larger accidents than would be implied by the direct product of probability times consequence.

It must be emphasized that the proposals for using probabilistic safety criteria were never aimed at performing *a posteriori* 'confirmatory' studies of the risk being incurred. Rather, they are aimed to check *a priori* that the prevention of accidents is coherent and consistent with the radiation protection principles. It should also be underlined that *a priori* probabilistic analysis allows firmly grounded anticipation, when there are frequency data that allow classical statistical treatment, and (with the help of Bayes's theorem) solidly founded inference when only professional judgement is available.

In sum, an approach to nuclear safety based on the radiation protection principles has a uniqueness: its coherence and consistency *vis-à-vis* both actual radiation safety situations and potential nuclear safety situations. This exceptionality is at the root of its claim that it is based on a common ethical approach.

11.7 Radiation safety standards

The epistemological basis provided by UNSCEAR and the radiation protection paradigm recommended by ICRP are converted into international radiation safety standards, for NPPs and other practices, under the aegis of the IAEA.

In performing its safety functions, the IAEA is contributing to what has been termed a *de facto* international *radiation safety regime* (González, 2004b, 2004c), which includes three key elements:

1. Legally binding international undertakings by States, usually in the form of safety-related international *conventions*
2. Globally agreed international safety *standards*
3. International provisions for facilitating the *application* of those standards.

11.7.1 International conventions

The legally binding international undertakings by States are, in legal language, international conventions. Under the auspices of the IAEA, four major radiation-safety related international conventions have been adopted in recent years, namely:

1. The Convention on Early *Notification* of a Nuclear Accident (IAEA, 1986b)
2. The Convention on *Assistance* in the Case of a Nuclear Accident or Radiological Emergency (IAEA, 1986c)
3. The Convention on *Nuclear Safety* (IAEA, 1994)
4. The Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management (the so-called '*Joint Convention*') (IAEA, 1997).

The obligations undertaken by signatory States of these Conventions apply *inter alia* to radiation protection of NPPs.

Another relevant undertaking for NPP operation is the *Radiation Protection Convention, 1960 (No. 115)* of the International Labour Organization (ILO, 1960). This Convention applies to all activities involving exposure of workers to ionizing radiations in the course of their work, including work at NPPs.

11.7.2 International standards

Pursuant to its Statute, the IAEA has established a body of standards in the fields of radiation safety, radioactive materials transport safety, radioactive waste safety, and nuclear safety. The standards follow a common general pattern – fundamental principles and a set of mandatory requirements – as follows:

1. *Safety Fundamentals*, stating basic objectives, concepts and principles
2. *Safety Requirements*, stating basic requirements, which must be fulfilled in the case of particular activities or applications
3. *Safety Guides*, containing recommendations related to the fulfilment of the basic requirements stated in the Standards.

Safety Fundamentals and Safety Requirements require the approval of government delegates at the IAEA's Board of Governors. Safety Guides are issued under the authority of the IAEA's Director General. A separate series of documents, the Safety Reports, gives examples and detailed descriptions of methods that can be applied in implementing the Standards.

The *Safety Fundamentals* (IAEA, 2006b) is the policy document of the IAEA safety standards, stating the basic objectives, concepts and principles involved in ensuring protection and safety in the development and application of atomic energy for peaceful purposes. They thereby provide the rationale for such activities having to fulfil certain requirements but do not state what those requirements are or provide technical details and generally do not discuss the application of principles. The formulation of some of the international Fundamental Safety Principles is based on the radiation protection principles. Currently there are 10 Fundamental Safety Principles, namely responsibility for safety; role of government; leadership and management for safety; justification of facilities and activities; optimization of protection; limitation of risks to individuals; protection of present and future generations; prevention of accidents; emergency preparedness and response; and protective actions to reduce existing or unregulated radiation risks. Four of them were extracted from the radiation protection principles, and are formulated as follows: justification of facilities and activities (facilities and activities that give rise to radiation risks must yield an overall benefit); optimization of protection (protection must be optimized to provide the highest level of safety that can reasonably be achieved); limitation of risks to individuals (measures for controlling radiation risks must ensure that no individual bears an unacceptable risk of harm); and protection of present and future generations (people and the environment, present and future, must be protected against radiation risks). The current Fundamentals are co-sponsored by six international organizations. They explain the fundamental basis for the approaches to protection and safety for those at senior levels in government and regulatory bodies, and for NPP operators, who may not be specialists in radiation protection and safety but who have decision-making responsibilities in such matters.

The *Safety Requirements* encompass the basic requirements that must be satisfied to ensure safety for particular activities or application areas. These requirements are governed by the basic objectives, concepts and principles presented in the Safety Fundamentals. The publications in this category do not present recommendations on, or explanations of, how to meet the requirements. The written style used in the Safety Requirements accords with that of regulatory documents since the requirements established may be adopted by Member States, at their own discretion, for use in national regulations. Regulatory requirements are expressed as 'shall' statements, are self-standing and do not cite standards of other organizations over which the IAEA has no control. They also are published in all official languages of the IAEA.

The *Safety Guides* encompass recommendations, based on international experience, of measures to ensure the observance of the Safety Requirements. Recommendations in the Safety Guides are expressed as 'should' statements and are issued under the authority of the Director General. A large number of Safety Guides support the Safety Requirements.

11.7.3 Providing for the application of international standards

In order to meet its second responsibility – to provide for the application of its standards – the IAEA carries out a number of safety-related activities. These include fostering information exchange, encouraging research and development, providing technical assistance to developing Member States, promoting education and training and rendering a number of safety services, such as radiological assessments of contaminated environments, the evaluation of accidents, and radiation protection appraisals carried out by international peers. In addition, any Member State may request the assistance of the IAEA in setting up a project involving nuclear technology and, before approving the project, the IAEA's Board of Governors is required to give due consideration to 'the adequacy of proposed health and safety standards . . .'. The IAEA is also responsible for international nuclear safeguards and – with respect to any IAEA project, or other arrangement where the IAEA is requested by the parties concerned to apply safeguards – has the right and responsibility 'to require the observance of any health and safety measure prescribed by the IAEA' and 'to send into the territory of the recipient State or States inspectors . . . to determine whether there is compliance with [such] health and safety measures.'

11.7.4 Main radiation safety requirements for NPPs

Among the international intergovernmental organizations involved in radiation safety, the IAEA is the only one specifically authorized under the terms of its Statute to establish radiation safety standards. Unsurprisingly the first endeavour to establish international radiation protection requirements was made at the IAEA, and has become the main international radiation safety requirement for all activities involving radiation exposure, including NPPs. Over time it has come to be known as 'basic safety standards', or BSS.

The IAEA's Board of Governors first approved radiation protection and safety 'measures' in March 1960 (IAEA, 1960, 1976), when it was stated that 'the IAEA's basic safety standards . . . will be based, to the extent possible, on the recommendations of the International Commission on Radiological Protection (ICRP).' The IAEA's Board of Governors first approved basic safety standards in June 1962, and these were published as Safety Series No. 9 (IAEA, 1962), a revised version being published in 1967 (IAEA, 1967). At the beginning of the 1980s a further – comprehensive – revision was carried out. This was jointly sponsored by the IAEA and two other organizations of the UN family, ILO and WHO, and also by the Nuclear Energy Agency of the Organization for Economic Cooperation and Development (OECD/NEA). The resulting text was published by the IAEA as the 1982 edition of Safety Series No. 9 (IAEA, 1982). At the end of the 1980s, the ICRP revised its standing advice and issued its 1990 recommendations (ICRP, 1991) in the light of which relevant organizations of the UN family and other multinational agencies promptly started to review their own radiation safety standards. Thus, taking account of the new developments, the IAEA, FAO, ILO, OECD/NEA, PAHO and WHO established a Joint Secretariat for the preparation of new *International Basic Safety Standards for Protection Against Ionizing Radiation and for the Safety of Radiation Sources*, which came to be commonly referred to as the Basic Safety Standards (or BSS) (IAEA, 1996a; González, 1994, 2001a). At the moment of preparation of this book the BSS are freshly revised (IAEA, 2011) to take account of the new ICRP recommendations (ICRP, 2007a).

For the particular case of NPPs, the BSS are supported by requirements on safe siting (IAEA, 2003a), design (IAEA, 2000c) and operation (IAEA, 2000d), which, *mutatis mutandi*, include radiation protection requirements. They are also sustained by a plethora of safety guides, including those on radiation protection aspects of design for nuclear power plants (IAEA, 2005c), on radiation protection and radioactive waste management in the operation of nuclear power plants (IAEA, 2002a) and on dispersion of radioactive material in air and water and consideration of population distribution in site evaluation (IAEA, 2002b).

11.8 Occupational protection at nuclear power plants (NPPs)

Occupational radiation protection at NPPs is internationally governed by the ILO Convention 115 (ILO, 1960), by the BSS (IAEA, 1996a, 2011) and by specific guidance on occupational radiation protection (IAEA, 1999a) and on assessment of occupational exposure due to intakes of radionuclides (IAEA, 1999b) and to external sources (IAEA, 1999c). These are fully based on specific ICRP recommendations (ICRP, 1997b). A wide international consensus exists in this area (IAEA, 2003b) and its international regulation (González, 2003b).

In short, the international accord establishes that all those persons engaged in work at NPPs are in principle considered occupational exposed workers, although the occupational protection standards apply *in toto* only to those performing work in controlled areas. Those organizations that employ them should be considered as employers. Both workers and employers should be subjected to responsibilities established in the occupational radiation protection normative.

Employers shall be responsible for protecting the workers and complying with any relevant requirements of the occupational radiation protection standards, ensuring in particular that the occupational exposures be limited as specified in the relevant requirements and that occupational protection and safety be optimized in accordance with the relevant requirements.

Employers should also ensure that decisions regarding measures for occupational protection and safety be recorded and made available to the workers through their representatives where appropriate. They should establish policies, procedures and organizational arrangements for protection and safety for implementing the relevant requirements, with priority given to measures for controlling occupational exposures.

Employers are also responsible for providing the following:

1. Suitable and adequate facilities, equipment and services for protection and safety, the nature and extent of which are commensurate with the expected magnitude and likelihood of the occupational exposure
2. Necessary health surveillance and health services, providing appropriate protective devices and monitoring equipment and arranging for its proper use
3. Suitable and adequate human resources and appropriate training in protection and safety, as well as periodic retraining and updating as required in order to ensure the necessary level of competence, keeping records of the training provided to individual workers
4. Adequate records of occupational exposure
5. Consultation and cooperation with workers with respect to protection and safety, concerning all measures necessary to achieve the effective implementation of requirements
6. Necessary conditions to promote a safety culture
7. In consultation with workers, writing rules and procedures as are necessary to ensure adequate levels of protection and safety, including values of any relevant dose level that require investigation or specific authorization and the procedure to be followed in the event that any such value is exceeded, and making such rules and procedures and the protective measures and safety provisions known to those workers to whom they apply
8. Supervision of any work involving occupational exposure and taking all reasonable steps to ensure that the rules, procedures, protective measures and safety provisions be observed
9. For all workers, adequate information on the health risks due to their occupational exposure, adequate instruction and training on protection and safety, and adequate information on the significance for protection and safety of their actions
10. For female workers, appropriate information on (i) the risk to the embryo or foetus due to exposure of a pregnant worker; (ii) the importance for a female worker of notifying her employer as soon as she

suspects that she is pregnant; and (iii) the risk to an infant ingesting radioactive substances by breast feeding.

Employers should ensure that workers exposed to radiation from sources that are not directly related to their work receive the same level of protection as if they were members of the public. They should obtain, as a precondition for engagement of workers, the previous occupational exposure history of such workers and other information as may be necessary to provide protection and safety.

They should also be transparent with the information. In fact they should take such administrative actions as are necessary to ensure that workers are informed that protection and safety are integral parts of a general occupational health and safety programme in which they have certain obligations and responsibilities for their own protection and the protection of others, and in particular record any report received from a worker that identifies circumstances which could affect compliance, and shall take appropriate action.

As far as recording is concerned, employers should arrange for the assessment of the occupational exposure of workers, on the basis of individual monitoring where appropriate, and ensure that adequate arrangements be made with appropriate dosimetry services under an adequate quality assurance programme. They should also arrange for appropriate health surveillance based on the general principles of occupational health and designed to assess the initial and continuing fitness of workers for their intended tasks. Finally, they should maintain exposure records for each worker, which shall include (1) information on the general nature of the work in the response involving occupational exposure; (2) information on doses, exposures and intakes at or above the relevant recording levels and the data upon which the dose assessments have been based; (3) when a worker is or has been occupationally exposed while in the employ of more than one employer, information on the dates of employment with each employer and the doses, exposures and intakes in each such employment; and (4) records of any doses, exposures or intakes due to other emergency interventions or accidents, as well as providing for access by workers to information in their own exposure records and for access to the exposure records by the supervisor of the health surveillance programme, facilitating the provision of copies of workers' exposure records to new employers when workers change employment, and preserving such records during the worker's working life and afterwards at least until the worker attains or would have attained the age of 75 years, and for not less than 30 years after the termination of the work involving occupational exposure.

On their side, workers shall be responsible for following any applicable rules and procedures for protection and safety specified by the employer and using properly the monitoring devices and the protective equipment and clothing provided. They should cooperate with the employer with respect to protection and safety and the operation of radiological health surveillance and dose assessment programmes and provide to the employer such information on their past and current work as is relevant to ensure effective and comprehensive protection and safety for themselves and others.

Workers should abstain from any wilful action that could put themselves or others in situations that contravene the requirements. The should accept such information, instruction and training concerning protection and safety as will enable them to conduct their work in accordance with the requirements of occupational radiation protection standards. Finally, they should be reporting to the employer, as soon as feasible, circumstances that could adversely affect compliance with the standards, if for any reason a worker is able to identify such circumstances.

It is interesting to note that according to the international labour normative, conditions of service of workers shall be independent of the existence or the possibility of occupational exposure. Special compensatory arrangements or preferential treatment with respect to salary or special insurance coverage, working hours, length of vacation, additional holidays or retirement benefits shall neither be granted nor be used as substitutes for the provision of proper protection and safety measures to ensure compliance with the requirements of the relevant occupational radiation protection standards.

As indicated heretofore, a female worker should, on becoming aware that she is pregnant or if she is nursing, notify the employer in order that her working conditions may be modified if necessary. The notification of pregnancy or nursing shall not be considered a reason to exclude a female worker from work; however, the employer of a female worker who has notified pregnancy or nursing shall adapt the working conditions in respect of occupational exposure so as to ensure that the embryo or fetus, or the nursing infant, is afforded the same broad level of protection as required for members of the public. Taking account the above requirements and the unavoidable uncertainties surrounding accident-response measures, in practice it might be unfeasible to occupy female workers in those conditions as emergency responders undertaking life-saving or other urgent actions. Under these circumstances, employers shall make every reasonable effort to provide such potential workers with suitable alternative employment.

The general dose limits for occupational exposure have been described before. In more detail, the 'normal' occupational exposure of any worker shall be so controlled that more of the following limits be exceeded:

- An effective dose of 20 mSv per year averaged over five consecutive years
- An effective dose of 50 mSv in any single year
- An equivalent dose to the lens of the eye of 150 mSv in a year
- An equivalent dose to the extremities (hands and feet) or the skin of 500 mSv in a year. (The equivalent dose limits for the skin apply to the average dose over 1 cm^2 of the most highly irradiated area of the skin. Skin dose also contributes to the effective dose, this contribution being the average dose to the entire skin multiplied by the tissue weighting factor for the skin)
- In special circumstances, the values for the single-year effective dose can be duplicated.

For 'abnormal' situations that may occur if an accident happens at an NPP, special conditions might be employed for *volunteers* engaged in recovery operations. For workers undertaking rescue operations that involve saving life, no dose restrictions are recommended in principle if, and only if, the benefit to others clearly outweighs the rescuer's own risk. Otherwise, for rescue operations involving the prevention of serious injury or the development of catastrophic conditions, every effort should be made to avoid deterministic effects on health – by keeping effective doses below 1000 mSv to avoid serious deterministic health effects, or below 500 mSv to avoid other prompt health effects (the latter criterion leaves a margin for error in avoiding deterministic effects because of the possible difficulty in determining the exact exposure conditions immediately after an unexpected abnormal situation and the possibility that the workers concerned may not have the level of training or experience usually required for responding to such an unexpected situation). For workers undertaking other immediate and urgent rescue actions to prevent injuries or large doses to many people, all reasonable efforts should be made to keep doses below 100 mSv of effective dose.

For emergency actions undertaken by workers engaged in recovery operations, the doses received should be treated as part of normal occupational exposure and the 'normal' occupational dose limits apply, namely a limit on effective dose of 20 mSv/year, averaged over five years (100 mSv in five years), with the further provision that the effective dose should not exceed 50 mSv in any single year, and annual equivalent dose limits of 150 mSv for the lens of the eye, 500 mSv for the skin (average dose over 1 cm^2 of the most highly irradiated area of the skin), and 500 mSv for the hands and feet.

It should be re-emphasized that those rescuers undertaking actions in which the dose may exceed 100 mSv of effective dose should be volunteers, and should be well prepared for dealing with the aftermath of a radiation emergency, i.e., they should be clearly and comprehensively informed in advance of the associated health risk and, to the extent feasible, be trained in the actions that may be required, including the use of protective measures.

11.9 Public protection at nuclear power plants (NPPs): controlling discharges into the environment

The radiation protection of the public at NPPs is governed by the undertakings in the Joint Convention and by the requirements in the BSS.

International guidance is available for the regulatory control of radioactive discharges to the environment (IAEA, 2000e) and for the environmental and NPPs for purposes of radiation protection (IAEA, 2005a).

In short, the public affected by NPP operations is protected by the control of discharges of radionuclides to the environment. International standards provide regulatory bodies with a structured approach to the limitation of such discharges from NPP operations and optimization of protection from such operations, which may be adapted to the specific legal and regulatory infrastructure within which such a body operates. They also give guidance on the responsibilities of operating organizations in conducting radioactive discharge operations.

Past experience demostrates that operational discharges are low and radiation exposure to the public from NPP operations has been minute. Figure 11.2 has shown how effective the regulatory instruments for limiting discharges from NPPs into the environment have been.

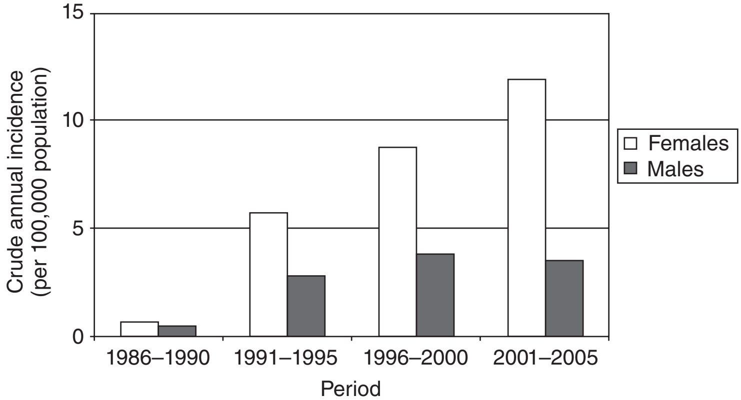
However, there is always the possibility, however remote, that a massive release of radioactive materials into the environment occurs as a result of a catastrophic accident. This is what happened as a result of the Chernobyl accident, a controversial topic that will close this chapter.

11.9.1 The Chernobyl accident

The 1986 accident at the Chernobyl nuclear power plant in the former Soviet Union was the most severe such accident in the history of civilian nuclear power and due to its perceived radiation consequences has become a nemesis for NPPs. However, while the accident undoubtedly was catastrophic in nature, and contaminated vast areas of European land, its radiation-related health consequences were fortunately limited, as can be observed in the maps of reference (De Cort *et al.*, 1998).

Since the fateful accident occurred, the international community has made unprecedented efforts to assess the magnitude and characteristics of its radiation-related health effects (IAEA, 1986a, 1988, 1991; González, 1990, 1996a, 1996b, 2007; Konstantinov and González, 1989). The results of those initiatives were synthesized at an international conference on the theme 'One decade after Chernobyl: summing up the consequences of the accident', which was held in Vienna in 1996 (IAEA, 1996b). Broadly similar conclusions were reached by the *Chernobyl Forum* launched by eight organizations of the United Nations system and the three most affected States to generate authoritative consensual statements on the environmental and health consequences attributable to radiation exposure and to provide advice on issues such as environmental remediation. The work of the Chernobyl Forum was appraised at an international conference on the theme 'Chernobyl: looking back to go forwards; towards a United Nations consensus on the effects of the accident and the future', which was held in Vienna in 2005 (IAEA, 2008a). The international consensus has been recently reported by UNSCEAR as follows (UNSCEAR, 2009):

1. A total of 134 plant staff and emergency workers received high doses of radiation that resulted in acute radiation syndrome (ARS), many of them also incurring skin injuries due to beta irradiation.
2. The high radiation doses proved fatal for 28 of those people in the first few months following the accident.
3. Although 19 ARS survivors had died by 2006, those deaths had different causes that usually were not associated with radiation exposure.
4. Skin injuries and radiation-related cataracts were among the main sequelae of ARS survivors.
5. Aside from the emergency workers, several hundred thousand people were involved in recovery operations but, apart from indications of an increase in incidence of leukaemia and of cataracts among those who received higher doses, there is to date no consistent evidence of health effects that can be attributed to radiation exposure.
6. A substantial increase in thyroid cancer incidence among persons exposed to the accident-related radiation as children or adolescents in 1986 has been observed in Belarus, Ukraine and four of the more affected regions of the Russian Federation. For the period 1991–2005, more than 6000 cases were reported, of which a substantial portion could be attributed to drinking milk in 1986 contaminated with iodine-131. Although thyroid cancer incidence continues to increase for this group, up to 2005 only 15 cases had proved fatal. Figure 11.8 presents the thyroid cancer incidence among people in Belarus who were children or adolescents at the time of the Chernobyl accident, for 1986–1990, 1991–1995, 1996–2000 and 2001–2005 (UNSCEAR, 2009).
7. Among the general public, to date there has been no consistent evidence of any other health effect that can be attributed to radiation exposure.



11.8 Thyroid cancer incidence among people in Belarus who were children or adolescents at the time of the Chernobyl accident.

In sum, based on 20 years of studies, UNSCEAR reconfirmed that, essentially, persons who were exposed as children to radioiodine from the Chernobyl accident and the emergency and recovery operation workers who received high doses of radiation are at increased risk of radiation-induced effects. *Most area residents were exposed to low-level radiation comparable to or a few times higher than the annual natural background radiation levels and need not live in fear of serious health consequences.*

Notwithstanding the above, it is clear that the Chernobyl accident has had and will continue to have an enormous impact on the development of nuclear energy and will be a continued prejudice in any assessment of its justification [González, 2007].

11.10 References and further reading

ARN (2010), Argentine Nuclear Regulatory Authority (Autoridad Regulatoria Nuclear). *Criterios Radiológicos Relativos a Accidentes en Reactores Nucleares de Potencia*; NORMA AR 3.1.3. Autoridad Regulatoria Nuclear, Buenos Aires, Argentina. http://www.arn.gob.ar/normas/3-1-3R2.pdf

Beninson, D. and Lindell, B. (1981), Critical views on the application of some methods for evaluation of accident probabilities and consequences, *Current Nuclear Power Plant Safety Issues*, Stockholm, 20–24 October 1980. Vienna, IAEA, 1981. Volume 2; pp. 325–340.

Beninson, D. (1996), Risk of radiation at low doses. The IRPA9 Sievert lecture. In: *Proceedings of the 1996 International Congress of Radiation Protection, Volume 1*. Vienna: Austrian Association of Radiation Protection: IRPA9.

Beninson, D. and González, A. J. (1981), Optimization of nuclear safety systems. In: *Proceedings of an International Conference on Current Nuclear Power Plants Safety Issues organized by the International Atomic Energy Agency*, Stockholm, 20–24 October 1980. Vienna, IAEA, 1981. Volume 2, pp. 449–456. IAEA-CN39/211 STI/PUB/566, IAEA, Vienna.

CAC (2006), Codex Alimentarius Commission. *Codex General Standard for Contaminants and Toxins in Foods*. CODEX STAN 193–1995, Rev. 2-2006. Codex Alimentarius Commission, Geneva.

CNEA (1979), Comisión Nacional de Energía Atómica. *Criterios radiológicos relativos a accidentes*. Consejo Asesor para el Licenciamiento de Instalaciones Nucleares. Buenos Aires, CNEA, 1979. 2 pp. NORMA CALIN N° 3.1.3. CNEA, Buenos Aires.

CNEA (1980), Comisión Nacional de Energía Atómica. *Análisis de falias para la evaluacion de riesgos*. Consejo Asesor para el Licenciamiento de Instalaciones Nucleares Buenos Aires, CNEA, 1980. 1 p. NORMA CALIN N° 3.2.2. CNEA, Buenos Aires.

De Cort, M., Dubois, G., Fridman, Sh. D., Germenchuk, M. G., Izrael, Yu. A., Janssens, A., Jones, A. R., Kelly, G. N., Kvasnikova, E. V., Matveenko, I. I., Nazarov, I. M., Pokumeiko, Yu. M., Sitak, V. A., Stukin, E. D., Tabachny, L. Ya., Tsaturov, Yu. S. and Avdyushin, S. I. (1998), *Atlas of Caesium Deposition on Europe after the Chernobyl Accident*. Luxembourg: Office for Official Publications of the European Communities; EUR Report 16733.

Edwards, A. A. and Lloyd, D. C. (1966), Risk from deterministic effects of ionising radiation. *Doc. NRPB*, Vol. 7, No. 3.

Gentner, N. and González, A. J. (2003), UNSCEAR's contribution to occupational radiation rrotection. In: *Occupational Radiation Protection: Protecting Workers against Exposure to Ionizing Radiation*, Proceedings of the International Conference organized by the International Atomic Energy Agency, etc., Geneva, 26–30 August 2002. Proceedings series, ISSN 0074-1884, STI/PUB/1145, ISBN 92-0-105603-6, pp. 69–86, IAEA, Vienna.

González, A. J. (1974), Un criterio para la evaluacion de la seguridad nuclear. In: *Proceedings of Symposium on Siting of Nuclear Facilities*, jointly organized by the International Atomic Energy Agency and the OECD Nuclear Energy Agency, Vienna, 9–13 December 1974. pp. 265–281. IAEA-SM-188/52, STI/PUB/384, IAEA, Vienna.

González, A. J. (1982), The regulatory use of probabilistic safety analysis in Argentina. *International Meeting on Thermal Reactor Safety*, Chicago, USA, 29 August–2 September 1982. NUREG-0027, Vol, 1.

González, A. J. (1986), The regulatory use of probabilistic safety analysis in Argentina. *Technical Committee on Status, Experience and Future Prospects for the Development of Probabilistic Safety Criteria*, Vienna, 27–31 January 1986. IAEA-TECDOC-524, IAEA, Vienna.

González, A. J. (1990), Recovery operations after the Chernobyl accident: the intervention criteria of the USSR's National Commission on Radiation Protection. In: *Proceedings of International Symposium on Recovery Operations in the Event of a Nuclear Accident or Radiological Emergency*, Vienna. IAEA SM-316157, p. 313.

González, A. J. (1994), Radiation safety: New international standards – the forthcoming International Basic Safety Standards for Protection against Ionizing Radiation and for the Safety of Radiation Sources are the product of unprecedented co-operation. *IAEA Bulletin*, 36(2), pp. 2–11, IAEA, Vienna.

González, A. J. (1996a), Chernobyl – Ten years after: Global experts clarify the facts about the 1986 accident and its effects. *IAEA Bulletin*, 38(3), pp. 2–13, IAEA, Vienna.

González, A. J. (1996b), IAEA Updating Report on the 'International Chernobyl Project' and the project 'One Decade after Chernobyl: Environmental Impact Assessment'. In: *Proceedings of the International Conference on One Decade after Chernobyl: Summing Up the Consequences of the Accident*, jointly sponsored by the European Commission, International Atomic Energy Agency, World Health Organization, in co-operation with the United Nations (Department of Humanitarian Affairs), etc., Vienna, 8–12 April 1996. – Proceedings series, ISSN 0074-1884, STI/PUB/1001, ISBN 92-0-103796-1, IAEA, Vienna.

González, A. J. (1998), Radiation and nuclear safety. *IAEA Bulletin*, 40(2), pp. 2–4, IAEA, Vienna.

González, A. J. (1999a), Timely action: Strengthening the safety of radiation sources and the security of radioactive materials, *IAEA Bulletin*, 41(3), pp. 2–17, IAEA, Vienna.

González, A. J. (1999b), International standards on the safety of radiation sources and the security of radioactive materials. In: *Proceedings of an International Conference on the Safety of Radiation Sources and the Security of Radioactive Materials* jointly organized by the European Commission etc., Dijon, France, 14–18 September 1998. IAEA-CN-70/B2.1. Proceedings series, ISSN 0074-1884, STI/PUB/1042, ISBN 92-0-101499-6, pp. 55–60. IAEA, Vienna.

González, A. J. (2000), The IAEA's policies on the safety of radioactive waste management. In: *Safety of Radioactive Waste Management: Proceedings of an International Conference* organized by the International Atomic Energy Agency etc., Cordoba, Spain, 13–17 March 2000. Proceedings series, ISSN 0074-1884, STI/PUB/1094, ISBN 92-0-101700-6, pp. 19–25. IAEA, Vienna.

González, A. J. (2001a), Décisions concernant l'exposition chronique du public aux rayonnements: Nouvelles recommandations de la CIPR. *Radioprotection*, 36(2), pp. 139–154. DOI: 10.1051/radiopro:2001113. O EDP Sciences. 2001 139.

González, A. J. (2001b), Security of radioactive sources: the evolving new international dimensions. *IAEA Bulletin*, 43(4), pp. 39–48, IAEA, Vienna.

González, A. J. (2002), The debate on the health effects attributable to low radiation exposure, 1 Pierce L. Rev. 39 (Copyright © 2002 Franklin Pierce Law Center), *Pierce Law Review*, Fall, 2002.

González, A. J. (2003a), Security of radioactive sources: Threats and answers. In: *Security of Radioactive Sources: Proceedings of an International Conference* organized by the International Atomic Energy Agency etc., Vienna, 10–13 March 2003. Proceedings series, ISSN 0074-1884, STI/PUB/1165, ISBN 92-0-107403-4, pp. 33–59. IAEA, Vienna.

González, A. J. (2003b), Occupational radiation protection: IAEA functions and policies. In: *Occupational Radiation Protection: Protecting Workers against Exposure to Ionizing Radiation, Proceedings of the International Conference* organized by the International Atomic Energy Agency, etc., Geneva, 26–30 August 2002. Proceedings series, ISSN 0074-1884, STI/PUB/1145, ISBN 92-0-105603-6, pp. 33–49. IAEA, Vienna.

González, A. J. (2003c), The Safety of radioactive waste management: Towards an international regime. In: *Issues and Trends in Radioactive Waste Management: Proceedings of an International Conference* organized by the International Atomic

Energy Agency in cooperation with the European Commission and the OECD Nuclear Energy Agency, hosted by the IAEA, Vienna, 9–13 December 2002. Proceedings series, ISSN 0074-1884, STI/PUB/1175, ISBN 92-0-113103-8, pp. 15–48. IAEA, Vienna.

González, A. J. (2003d), Safe nuclear decommissioning: Need for an international common approach. In: *Safe Decommissioning for Nuclear Activities, Proceedings of the International Conference* organized by the International Atomic Energy Agency and hosted by the Government of Germany through the Bundesamt für Strahlenschutz, Berlin, 14–18 October 2002. Proceedings series, ISSN 0074-1884, STI/PUB/1154, ISBN 92-0-109703-4, pp. 11–24. IAEA, Vienna.

González, A. J. (2004a), The IAEA and the safe transport of radioactive material. In: *Proceedings of an International Conference on the Safety of Transport of Radioactive Material*, organized by the International Atomic Energy Agency, co-sponsored by the International Civil Aviation Organization etc., Vienna, 7–11 July 2003. Proceedings series, ISSN 0074-1884, STI/PUB/1200, ISBN 92-0-108504-4, pp. 49–59. IAEA, Vienna.

González, A. J. (2004b), Radiation safety standards and their application: International policies and current issues. *Health Physics*, 87(3), pp. 258–272. doi: 10.1097/01.HP.0000130400.90548.5e. 2004.

González, A. J. (2004c), Protecting life against the detrimental effects attributable to radiation exposure: Towards a globally harmonized radiation protection regime. The Sievert Lecture, at the 11th International Congress of the International Radiation Protection Association, Spain, 23–28 May 2004.

González, A. J. (2005), Protecting the environment from radiation exposure: International efforts for restricting discharges of radioactive substances. In: *Protection of the Environment from the Effects of Ionizing Radiation: Proceedings of an International Conference*, Stockholm, 6–10 October 2003. Proceedings series, ISSN 0074-1884, STI/PUB/1229, ISBN 92-0-104805-X, pp. 87–97. IAEA, Vienna.

González, A. J. (2006), The road from Dijon to Bordeaux. In: *Safety and Security of Radioactive Sources: Towards a Global System for the Continuous Control of Sources Throughout their Life Cycle: Proceedings of an International Conference* organized by the International Atomic Energy Agency in cooperation with the European Commission etc., Bordeaux, France, 27 June–1 July 2005, Proceedings series, ISSN 0074-1884, STI/PUB/1262, ISBN 92-0-108306-8. IAEA, Vienna.

González, A. J. (2007), Chernobyl vis-à-vis the nuclear future: An international perspective. *Health Physics*, 93(5), pp. 571–592, November 2007. doi: 10.1097/01.HP.0000282037.88438.3d. 2007_Chernobyl_HealthPhys_Vol93(5)_p571-592_2007[1].pdf

González, A. J. (2009a), The 12th Congress of the International Radiation Protection Association: Strengthening Radiation Protection Worldwide. *Health Physics*, 97(1). pp. 6–49, July 2009. doi: 10.1097/01.HP.0000348021.31830.54

González, A. J. (2009b), International approaches to remediation of territorial radioactive contamination. In: *Radioactivity in the Environment*, M.S. Baxter, editor, Vol 14, *Remediation of Contaminated Environments*, G. Voigt and S. Fesenko, editors. Amsterdam: Elsevier, pp. 1–40. ISBN 978-0-08-044862-6.

González, A. J. (2010), The Argentine approach to radiation safety: Its ethical basis. *Science and Technology of Nuclear Installations*, issue on Nuclear Activities in Argentina, 2010 (NAIA). Manuscript no. 910718.v2 (review article), http://mts.hindawi.com/author/910718/

IAEA (1960), International Atomic Energy Agency. *The IAEA's health and safety measures*. INFCIRC/18, IAEA, Vienna.

IAEA (1962), International Atomic Energy Agency. *Basic safety standards for radiation protection.* Safety Series No. 9, IAEA, Vienna.

IAEA (1967), International Atomic Energy Agency. *Basic safety standards for radiation protection (1967 edition).* Safety Series No. 9, IAEA, Vienna.

IAEA (1976), International Atomic Energy Agency. *The IAEA's health and safety measures*. INFCIRC/18/Rev. 1, IAEA, Vienna.

IAEA (1982), International Atomic Energy Agency. *Basic safety standards for radiation protection ([revised]1982 edition*). Safety Series No. 9, IAEA, Vienna.

IAEA (1986a), International Atomic Energy Agency. International Nuclear Safety Advisory Group. *Summary report on the post-accident review meeting on the Chernobyl accident*. Safety Series No. 75-INSAG-I, IAEA, Vienna.

IAEA (1986b), International Atomic Energy Agency. *Convention on Early Notification of a Nuclear Accident.* INFCIRC/335, IAEA, Vienna.

IAEA (1986c), International Atomic Energy Agency. *Convention on Assistance in the Case of a Nuclear Accident or Radiological Emergency.* INFCIRC/336, IAEA, Vienna.

IAEA (1988), International Atomic Energy Agency. *Proceedings of the All-Union Conference on the Medical Aspects of the Chernobyl Accident*. TECDOC 516, IAEA, Vienna.

IAEA (1991), International Atomic Energy Agency. *The international Chernobyl project: assessment of radiological consequences and evaluation of protective measures.* IAEA, Vienna.

IAEA (1994), International Atomic Energy Agency. *Convention on nuclear safety*. INFCIRC/449, IAEA, Vienna.

IAEA (1996a), International Atomic Energy Agency. *International basic safety standards for protection against ionizing radiation and for the safety of radiation sources.* Safety series, ISSN 0074-1892; no. 115 Safety standards. STI/PUB/996, ISBN 92-0-104295-7, IAEA, Vienna.

IAEA (1996b), International Atomic Energy Agency. *Proceedings of the International Conference on One Decade After Chernobyl: Summing Up the Consequences of the Accident*, jointly sponsored by the European Commission, International Atomic Energy Agency, World Health Organization, in co-operation with the United Nations (Department of Humanitarian Affairs) etc., Vienna, 8–12 April 1996. Proceedings series, ISSN 0074-1884, STI/PUB/1001, ISBN 92-0-103796-1. IAEA, Vienna.

IAEA (1997), International Atomic Energy Agency. *Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management*. INCIRC/546, IAEA, Vienna.

IAEA (1999a), International Atomic Energy Agency. *Occupational radiation protection. Safety guide*. Series No. RS-G-1.1, IAEA, Vienna.

IAEA (1999b), International Atomic Energy Agency. *Assessment of occupational exposure due to intakes of radionuclides. Safety guide*. Series No. RS-G-1.2, IAEA, Vienna.

IAEA (1999c), International Atomic Energy Agency. *Assessment of occupational exposure due to external sources of radiation. Safety guide*. Series No. RS-G-1.3, IAEA, Vienna.

IAEA (2000a), International Atomic Energy Agency. *Proceedings of an International Conference on the Safety of Radiation Sources and the Security of Radioactive Materials* jointly organized by the European Commission etc., Dijon, France, 14–18 September 1998. Proceedings series, IAEA-CN-70/B2.1, ISSN 0074-1884, STI/PUB/1042, ISBN 92-0-101499-6. IAEA, Vienna.

IAEA (2000b), International Atomic Energy Agency. *Safety of Radioactive Waste Management*: Proceedings of an International Conference organized by the International Atomic Energy Agency etc., Cordoba, Spain, 13–17 March 2000. Proceedings series, ISSN 0074-1884, STI/PUB/1094, ISBN 92-0-101700-6, IAEA, Vienna.

IAEA (2000c), International Atomic Energy Agency. *Safety of nuclear power plants: Design safety requirements.* Series No. NS-R-1, IAEA, Vienna.

IAEA (2000d), International Atomic Energy Agency. *Safety of nuclear power plants: Operation safety requirements*. Series No. NS-R-2, IAEA, Vienna.

IAEA (2000e), International Atomic Energy Agency. *Regulatory control of radioactive discharges to the environment. Safety guide*. Series No. WS-G-2.3, IAEA, Vienna.

IAEA (2001), International Atomic Energy Agency. *National Regulatory Authorities with Competence in the Safety of Radiation Sources and the Security of Radioactive Materials*. Proceedings of the Conference organized by the International Atomic Energy Agency and hosted by the Government of Argentina, Buenos Aires, Argentina, 11–15 December 2000. IAEA-CSP-9/P, ISSN 1563-0153. IAEA, Vienna.

IAEA (2002a), International Atomic Energy Agency. *Radiation protection and radioactive waste management in the operation of nuclear power plants. Safety guide*. Series No. NS-G-2.7, IAEA, Vienna.

IAEA (2002b), International Atomic Energy Agency. *Dispersion of radioactive material in air and water and consideration of population distribution in site evaluation for nuclear power plants. Safety guide*. Series No. NS-G-3.2, IAEA, Vienna.

IAEA (2002c), International Atomic Energy Agency. *Security of Radioactive Sources*. Proceedings of an International Conference organized by the International Atomic Energy Agency etc., Vienna, 10–13 March 2003. Proceedings series, ISSN 0074-1884, STI/PUB/1165, ISBN 92-0-107403-4. IAEA, Vienna.

IAEA (2003a), International Atomic Energy Agency. *Site evaluation for nuclear installations. Safety requirements.* Series No. NS-R-3, IAEA, Vienna.

IAEA (2003b), International Atomic Energy Agency. *Occupational Radiation Protection: Protecting Workers against Exposure to Ionizing Radiation*. Proceedings of the International Conference organized by the International Atomic Energy Agency etc., Geneva, 26–30 August 2002. Proceedings series, ISSN 0074-1884, STI/PUB/1145, ISBN 92-0-105603-6. IAEA, Vienna.

IAEA (2003c), International Atomic Energy Agency. *Safe Decommissioning for Nuclear Activities*. Proceedings of the International Conference organized by the International Atomic Energy Agency and hosted by the Government of Germany through the Bundesamt für Strahlenschutz, Berlin, 14–18 October 2002. Proceedings series, ISSN 0074-1884, STI/PUB/1154, ISBN 92-0-109703-4, pp. 11–24. IAEA, Vienna.

IAEA (2003d), International Atomic Energy Agency. *Issues and Trends in Radioactive Waste Management*. Proceedings of an International Conference organized by the International Atomic Energy Agency in cooperation with the European Commission and the OECD Nuclear Energy Agency, hosted by the IAEA, Vienna, 9–13 December 2002. Proceedings series, ISSN 0074-1884, STI/PUB/1175, ISBN 92-0-113103-8, pp. 15–48. IAEA, Vienna.

IAEA (2004a), International Atomic Energy Agency. *Safety of Transport of Radioactive Material*. Proceedings of an International Conference organized by the International Atomic Energy Agency, co-sponsored by the International Civil Aviation Organization etc., Vienna, 7–11 July 2003. Proceedings series, ISSN 0074-1884, STI/PUB/1200, ISBN 92-0-108504-4, pp. 49–59. IAEA, Vienna.

IAEA (2004b), International Atomic Energy Agency. *Measures to strengthen international cooperation in nuclear radiation and transport safety and waste management.* Resolution of the IAEA General Conference GC(48)/RES/10 under 805 A, 4, pt. 23; Radiological Criteria for Radionuclides in Commodities. IAEA, Vienna.

IAEA (2004c), International Atomic Energy Agency. *Application of the concepts of exclusion, exemption and clearance. Safety guide*. RS-G-1.7, IAEA, Vienna.

IAEA (2005a), International Atomic Energy Agency. *Environmental and source monitoring for purposes of radiation protection. Safety guide.* Series No. RS-G-1.8, IAEA, Vienna.

IAEA (2005b), International Atomic Energy Agency. *Proceedings of an International Conference on Protection of the Environment from the Effects of Ionizing Radiation*, Stockholm, 6–10 October 2003. IAEA Proceedings series, ISSN 0074-1884, STI/PUB/1229, ISBN 92-0-104805-X. IAEA, Vienna.

IAEA (2005c), International Atomic Energy Agency. *Radiation protection aspects of design for nuclear power plants. Safety guide*. Series No. NS-G-1.13, IAEA, Vienna.

IAEA (2006a), International Atomic Energy Agency. *International Conference on Chernobyl: Looking Back to go Forward.* lAEA, Vienna.

IAEA (2006b), International Atomic Energy Agency. *Fundamental Safety Principles: Safety Fundamentals*. IAEA Safety Standards series, No. SF-1, ISSN 1020-525X, STI/PUB/1273, ISBN 92-0-110706-4. IAEA, Vienna.

IAEA (2006c), International Atomic Energy Agency. *Safety and Security of Radioactive Sources: Towards a Global System for the Continuous Control of Sources Throughout their Life Cycle*. Proceedings of an International Conference organized by the International Atomic Energy Agency in cooperation with the European Commission etc., Bordeaux, France, 27 June–1 July 2005. Proceedings series, ISSN 0074-1884, STI/PUB/1262, ISBN 92-0-108306-8. IAEA, Vienna.

IAEA (2007), International Atomic Energy Agency. *Decommissioning of Nuclear Facilities and the Safe Termination of Nuclear Activities*. Proceedings of the Conference organized by the International Atomic Energy Agency, co-sponsored by the European Commission, in cooperation with the OECD Nuclear Energy Agency and the World Nuclear Association, hosted by the Government of Greece through the Ministry of Foreign Affairs of the Hellenic Republic and the Greek Atomic Energy Commission, Athens, 11–15 December 2006. Proceedings series, ISSN 0074-1884, STI/PUB/1299, ISBN 978-92-0-106107-2, pp. 357–367, IAEA, Vienna.

IAEA (2008a), International Atomic Energy Agency. *Proceedings of the International Conference on Chernobyl: Looking Back to go Forward*, organized by the

International Atomic Energy Agency on behalf of the Chernobyl Forum, Vienna, 6–7 September 2005. Proceedings series, ISSN 0074-1884, STI/PUB/1312, ISBN 978-92-0-110807-4 . IAEA, Vienna.

IAEA (2008b), International Atomic Energy Agency. *Advisory material for the IAEA regulations for the safe transport of radioactive material. Safety guide*. Series No. TS-G-1.1 (Rev.1), IAEA, Vienna.

IAEA (2009), International Atomic Energy Agency. *Regulations for the safe transport of radioactive material*, 2009 edition. Series No. TS-R-1, IAEA, Vienna.

IAEA (2010), International Atomic Energy Agency. *IRPA12: 12th Congress of the International Radiation Protection Association (IRPA): Strengthening Radiation Protection Worldwide: Highlights, Global Perspective and Future Trends*, Buenos Aires, Argentina, 19–24 October 2008. Proceedings series, ISSN 0074-1884, STI/PUB/1460, ISBN 978-92-0-105410-4. IAEA, Vienna.

IAEA (2011), International Atomic Energy Agency. *International Basic Safety Standards for Protection against Ionizing Radiation and for the Safety of Radiation Sources*. Safety Requirements DS379. Cosponsored by the Food and Agriculture Organization of the United Nations, International Atomic Energy Agency, International Labour Organization, Nuclear Energy Agency of the Organization for Economic Co-operation and Development, Pan American Health Organization, World Health Organization, and potentially by the European Commission and the United Nations Environment Programme. IAEA (in preparation).

ICRP (1951), International Commission on Radiological Protection. *International Recommendations on Radiological Protection.* Revised by the International Commission on Radiological Protection and the 6th International Congress of Radiology, London, 1950. *Br. J. Radiol.*, 24, 46–53.

ICRP (1955), International Commission on Radiological Protection. *Recommendations of the International Commission on Radiological Protection. Br. J. Radiol.* (Suppl. 6).

ICRP (1957), International Commission on Radiological Protection. *Reports on Amendments during 1956 to the Recommendations of the International Commission on Radiological Protection* (ICRP). *Acta Radiol.*, 48, 493–495.

ICRP (1959), International Commission on Radiological Protection. *Recommendations of the International Commission on Radiological Protection*. ICRP Publication 1. Pergamon Press, Oxford, UK.

ICRP (1964), International Commission on Radiological Protection. *Recommendations of the International Commission on Radiological Protection*. ICRP Publication 6. Pergamon Press, Oxford, UK.

ICRP (1966), International Commission on Radiological Protection. *Recommendations of the International Commission on Radiological Protection*. ICRP Publication 9, Pergamon Press, Oxford, UK.

ICRP (1973), International Commission on Radiological Protection. *Implications of Commission Recommendations that doses be kept as low as readily achievable*. ICRP Publication 22. Pergamon Press, Oxford, UK.

ICRP (1977), International Commission on Radiological Protection. *Recommendations of the International Commission on Radiological Protection*. ICRP Publication 26, *Ann. ICRP*, 1(3).

ICRP (1978), International Commission on Radiological Protection. *Statement from the 1978 Stockholm meeting of the ICRP*. ICRP Publication 28, *Ann. ICRP*, 2(1).

ICRP (1980), International Commission on Radiological Protection. *Cost-benefit analysis in the optimization of radiation protection*. ICRP Publication 37, *Ann. ICRP*, 10(2–3).

ICRP (1984), Nonstochastic effects of ionizing radiation. ICRP Publication 41. *Ann. ICRP*, 14(3).

ICRP (1985a), International Commission on Radiological Protection. *Quantitative bases for developing a unified index of harm*. ICRP Publication 45. Includes statement from the 1985 Paris meeting of the ICRP. *Ann. ICRP*, 15(3).

ICRP (1985b), International Commission on Radiological Protection. *Radiation protection principles for the disposal of solid radioactive waste*. ICRP Publication 46, *Ann. ICRP*, 15(4).

ICRP (1990), International Commission on Radiological Protection. *Optimization and decision making in radiological protection*. ICRP Publication 55, *Ann. ICRP*, 20(1).

ICRP (1991), International Commission on Radiological Protection. *1990 Recommendations of the International Commission on Radiological Protection.* ICRP Publication 60, Ann. ICRP, 21(1–3).

ICRP (1993), International Commission on Radiological Protection. *Protection from potential exposure: A conceptual framework*. ICRP Publication 64, *Ann. ICRP*, 23(1).

ICRP (1997a), International Commission on Radiological Protection. *Radiological protection policy for the disposal of radioactive waste.* ICRP Publication 77, *Ann. ICRP*, 27(Supplement).

ICRP (1997b), International Commission on Radiological Protection. *General principles for the radiation protection of workers.* ICRP Publication 75, *Ann. ICRP*, 27(1).

ICRP (1998), International Commission on Radiological Protection. *Radiation protection recommendations as applied to the disposal of long-lived solid radioactive waste.* ICRP Publication 81, *Ann. ICRP*, 28(4).

ICRP (1999), International Commission on Radiological Protection. *Protection of the public in situations of prolonged radiation exposure.* ICRP Publication 82, *Ann. ICRP*, 29(1–2).

ICRP (2003), International Commission on Radiological Protection. *A framework for assessing the impact of ionising radiation on non-human species.* ICRP Publication 91, *Ann. ICRP*, 33(3).

ICRP (2005a), International Commission on Radiological Protection. *Protecting people against radiation exposure in the event of a radiological attack*. ICRP Publication 96 (chairman of the Task Group), *Ann. ICRP*, 35(1).

ICRP (2005b), International Commission on Radiological Protection. *Low dose extrapolation of radiation-related cancer risk*. ICRP Publication 99, *Ann. ICRP*, 35(4).

ICRP (2006a), International Commission on Radiological Protection. *Assessing dose of the representative person for the purpose of radiation protection of the public.* ICRP Publication 101a. *Ann. ICRP*, 36(3) (published under the same cover as ICRP, 2006b).

ICRP (2006b), International Commission on Radiological Protection. *The optimisation of radiological protection – Broadening the process*. ICRP Publication 101b. *Ann. ICRP*, 36(3) (published under the same cover as ICRP, 2006a).

ICRP (2007a), International Commission on Radiological Protection. *The 2007 Recommendations of the International Commission on Radiological Protection.* ICRP Publication 103. *Ann. ICRP*, 37(2–4).

ICRP (2007b), International Commission on Radiological Protection. *Scope of radiological protection control measures*. ICRP Publication 104. *Ann. ICRP*, 37(5).

ICRP (2008), International Commission on Radiological Protection. *Environmental protection – the concept and use of reference animals and plants*. ICRP Publication 108. *Ann. ICRP*, 38(4–6).

ICRP (2009a), International Commission on Radiological Protection. *Application of the Commission's recommendations to the protection of people living in long-term contaminated areas after a nuclear accident or a radiation emergency*. ICRP Publication 111. *Ann. ICRP*, 39(3).

ICRP (2009b), International Commission on Radiological Protection. *Application of the Commission's recommendations for the protection of people in emergency exposure situations*. ICRP Publication 109. *Ann. ICRP*, 39(1).

ICRU (1938), International Commission on Radiation Units and Measurements. *Recommendations of the International Commission on Radiation Units*, Chicago, 1937. *Am. J. Roentgenol., Radium Therapy Nucl. Med.*, 39, 295.

ICRU (1954), International Commission on Radiation Units and Measurements. *Recommendations of the International Commission on Radiation Units*, Copenhagen, 1953. *Radiology*, 62, 106.

ICRU (1962), International Commission on Radiation Units and Measurements. *Radiation Quantities and Units*, Report 10a of the International Commission on Radiation Units and Measurements, *Natl. Bur. Std Handbook*, 78.

ILO (1960), International Labour Organization. *Radiation Protection Convention, 1960 (No. 115)*; General Conference of the International Labour Organization, Forty-fourth Session, 1 June 1960; International Labour Office, Geneva, 1960. (See also International Labour Organization, *Radiation Protection of Workers. Ionising Radiations – An ILO Code of Practice*, ISBN 92-2-105996-0, International Labour Office, Geneva).

ILO (2010), International Labour Organization. *Approaches to Attribution of Detrimental Health Effects to Occupational Ionizing Radiation Exposure and their Application in Compensation Programmes for Cancer.* Jointly published by the International Labour Organization (ILO), the International Atomic Energy Agency (IAEA) and the World Health Organization (WHO). Publication ILO-OSH 73, ISBN 978-92-2-122413-6. International Labour Office, Geneva.

IXRPC (1928), International X-ray and Radium Protection Committee. *X-ray and Radium Protection. Recommendations of the 2nd International Congress of Radiology*, 1928. *Br. J. Radiol.*, 12, pp. 359–363.

IXRPC (1934), International X-ray and Radium Protection Committee. *International Recommendations for X-ray and Radium Protection*. Revised by the International X-ray and Radium Protection Commission and adopted by the 4th International Congress of Radiology, Zurich, July 1934. *Br. J. Radiol.*, 7, 1–5.

Konstantinov, L. V. and González, A. J. (1989), The radiological consequences of the Chernobyl accident. *Nuclear Safety*, 30(1).

Scott, B. R. (1993), Early occurring and continuing effects. In: *Modification of Models Resulting from Addition of Effects of Exposure to Alpha-emitting Nuclides*.

Washington, DC, Nuclear Regulatory Commission, NUREG/CR-4214, Rev 1, Part II, Addendum 2 (LMF-136).

Scott, B. R. and Hahn, F. F. (1989), *Early occurring and continuing effects models for nuclear power plant accident consequence analysis. Low-LET radiation*. Washington, DC, Nuclear Regulatory Commission, NUREG/CR-4214 (SAND85-7185), Rev. 1, Part II.

UNESCO (2005), United Nations Educational, Scientific and Cultural Organization. *The Precautionary Principle*. UNESCO, Paris.

UNSCEAR (1958), United Nations. *Report of the United Nations Scientific Committee on the Effects of Atomic Radiation*. Official Records of the General Assembly, Thirteenth Session, Supplement No. 17 (A/3838). United Nations, New York.

UNSCEAR (1962), United Nations. *Report of the United Nations Scientific Committee on the Effects of Atomic Radiation*. Official Records of the General Assembly, Seventeenth Session, Supplement No. 16 (A/5216). United Nations, New York.

UNSCEAR (1964), United Nations. *Report of the United Nations Scientific Committee on the Effects of Atomic Radiation*. Official Records of the General Assembly, Nineteenth Session, Supplement No. 14 (A/5814). United Nations, New York.

UNSCEAR (1966), United Nations. *Report of the United Nations Scientific Committee on the Effects of Atomic Radiation*. Official Records of the General Assembly, Twenty-First Session, Supplement No. 14 (A/6314). United Nations, New York.

UNSCEAR (1969), United Nations. *Report of the United Nations Scientific Committee on the Effects of Atomic Radiation*. Official Records of the General Assembly, Twenty-Fourth Session, Supplement No. 13 (A/7613). United Nations, New York.

UNSCEAR (1972), United Nations. *Ionizing Radiation: Levels and Effects. Volume I: Levels; Volume II: Effects*. United Nations Scientific Committee on the Effects of Atomic Radiation, 1972 Report to the General Assembly, with annexes. United Nations sales publications E.72.IX.17 and 18. United Nations, New York.

UNSCEAR (1977), United Nations. *Sources and Effects of Ionizing Radiation*. United Nations Scientific Committee on the Effects of Atomic Radiation, 1977 Report to the General Assembly, with annexes. United Nations sales publication E.77.IX.1. United Nations, New York.

UNSCEAR (1982), United Nations. *Ionizing Radiation: Sources and Biological Effects*. United Nations Scientific Committee on the Effects of Atomic Radiation, 1982 Report to the General Assembly, with annexes. United Nations sales publication E.82.IX.8. United Nations, New York.

UNSCEAR (1986), United Nations. *Genetic and Somatic Effects of Ionizing Radiation*. United Nations Scientific Committee on the Effects of Atomic Radiation, 1986 Report to the General Assembly, with annexes. United Nations sales publication E.86.IX.9. United Nations, New York.

UNSCEAR (1988), United Nations. *Sources, Effects and Risks of Ionizing Radiation*. United Nations Scientific Committee on the Effects of Atomic Radiation, 1988 Report to the General Assembly, with annexes. United Nations sales publication E.88.IX.7. United Nations, New York.

UNSCEAR (1993), United Nations. *Sources and Effects of Ionizing Radiation*. United Nations Scientific Committee on the Effects of Atomic Radiation, 1993 Report to the General Assembly, with scientific annexes. United Nations sales publication E.94.IX.2. United Nations, New York.

UNSCEAR (1994), United Nations. *Sources and Effects of Ionizing Radiation*. United Nations Scientific Committee on the Effects of Atomic Radiation, 1994 Report to the General Assembly, with scientific annexes. United Nations sales publication E.94.IX.11. United Nations, New York.

UNSCEAR (1996), United Nations. *Sources and Effects of Ionizing Radiation*. United Nations Scientific Committee on the Effects of Atomic Radiation, 1996 Report to the General Assembly, with scientific annex. United Nations sales publication E.96.IX.3. United Nations, New York.

UNSCEAR (2000), United Nations. *Sources and Effects of Ionizing Radiation. Volume I: Sources; Volume II: Effects*. United Nations Scientific Committee on the Effects of Atomic Radiation, 2000 Report to the General Assembly, with scientific annexes. United Nations sales publications E.00.IX.3 and E.00.IX.4. United Nations, New York.

UNSCEAR (2001), United Nations. *Hereditary Effects of Radiation*. United Nations Scientific Committee on the Effects of Atomic Radiation, 2001 Report to the General Assembly, with scientific annex. United Nations sales publication E.01. IX.2. United Nations, New York.

UNSCEAR (2009), United Nations. *Effects of Ionizing Radiation. Volume I: Report to the General Assembly, Scientific Annexes A and B; Volume II: Scientific Annexes C, D and E.* United Nations Scientific Committee on the Effects of Atomic Radiation, UNSCEAR 2006 Report. United Nations sales publications E.08.IX.6 (2008) and E.09.IX.5 (2009). United Nations, New York.

UNSCEAR (2011), United Nations. *Sources and Effects of Ionizing Radiation. Volume I: Sources; Volume II: Sources and Effects* (to be published). United Nations Scientific Committee on the Effects of Atomic Radiation, UNSCEAR 2008 Report. United Nations sales publication E.10.XI.3. United Nations, New York.

WHO (2004), World Health Organization. *Guidelines for Drinking-water Quality*, 3rd edition. World Health Organization, Geneva.

12
Emergency planning in nuclear power programmes

E. GIL LÓPEZ, IAEA Radiation Safety Regulator, Austria

Abstract: Despite nuclear facilities being designed, constructed and operated according to the most stringent safety regulations, accidents, human failures, extreme external events or malicious acts can occur that require the implementation of adequate emergency actions. Since the Chernobyl accident in 1986, many efforts have been devoted to improving the nuclear emergency response at national and international levels, and emergency planning and preparedness have become a significant activity of the safety provisions needed to put in service a nuclear power plant. National regulations, usually based on international standards, establish the technical requirements for emergency planning and allocate responsibilities to plant operators and governmental bodies in charge of its implementation. Giving a suitable response to a nuclear accident requires efficient coordination among intervention organizations, emergency coordination centres are operated to facilitate such coordination, and regular exercises are performed to train intervention staff and improve emergency plans and procedures at every level.

Key words: emergency plans, emergency response, coordination centres, intervention organizations, international standards and recommendations.

12.1 Introduction

Nuclear and radiological emergencies can occur in a wide range of facilities, including fixed and mobile nuclear reactors; facilities for the mining and processing of radioactive ores; facilities for fuel reprocessing and other fuel cycle facilities; facilities for the management of radioactive waste; the trans-

---

Published by Woodhead Publishing Limited, 2012

port of radioactive material; sources of radiation used in industrial, agricultural, medical, research and teaching applications; facilities using radiation or radioactive material; and satellites and radio-thermal generators using radiation sources or reactors. The common characteristic of nuclear and radiological emergencies is that both involve hazards associated with ionizing radiation. In coherence with the rest of this book, this chapter is specifically aimed at emergency planning at nuclear power plants.

Nuclear facilities contain large amounts of nuclear material that can generate radioactive material by a chain reaction or by activation of stable nuclides that have been exposed to high neutron flux. Nuclear reactors can accumulate a large amount of radioactive materials, depending on their thermal power, the fuel burn-up and the time elapsed since the last shutdown. Multiple barriers contain these radioactive materials and prevent their radiation from damaging facility workers and the environment. Some critical components of a nuclear facility, such as the reactor core, need permanent cooling because radioactive decay of fission products generates a large amount of energy that could damage them if it is not extracted efficiently. An accident or an intentional action could disable the reactivity control systems, the cooling systems or the barriers containing radioactive materials. In this case, large amounts of these materials could escape to the environment. The energy accumulated within the facility can contribute to the spreading of radioactive materials into the environment over a wide area.

The fundamental safety objective in the use of nuclear and radiation techniques is to protect people and the environment from harmful effects of ionizing radiation. This objective has to be achieved without unduly limiting the operation of facilities or the conduct of activities that give rise to radiation risks. To reach this objective all reasonable efforts must be made to prevent nuclear or radiation accidents and mitigate their consequences.

The most harmful consequences arising from nuclear facilities and activities have come from loss of control over the nuclear reactor core, nuclear chain reaction or radioactive source. Consequently, in order to ensure that the likelihood of an accident having harmful consequences is extremely low, measures have to be taken:

- To prevent the occurrence of abnormal conditions, including breaches of security, that could lead to such a loss of control
- To prevent the escalation of any such failures or abnormal conditions that do occur
- To prevent the loss of control over radioactive sources.

Taking measures towards achieving these goals by undertaking interventions, which are defined as any action intended to reduce or avert exposure or the likelihood of exposure to sources which are not part of a controlled practice or which are out of control as a consequence of an accident, is governed at all times by the principles of justification and optimization recommended by the International Commission on Radiological Protection, ICRP (ICRP, 1991, 1993). According to the ICRP, any proposed intervention that does more good than harm is justified, and the form, scale and duration of any intervention shall be optimized so that the net benefit is maximized.

Every nuclear facility is designed to prevent any accident that can occur according to the applicable regulation. Two approaches are commonly used to demonstrate the compliance with regulation: the deterministic approach is used to demonstrate that the design is enough to prevent all regulated design-basis accidents and mitigate their consequences if they were to occur; the probabilistic methodology is used to verify that the accidents behind the design basis, that is the so-called severe accidents, should have a very low probability of occurrence and their consequences should be mitigated by dedicated design features. In addition, every nuclear facility has an emergency plan to be activated in case of an accident or malicious act to prevent severe damage to the facility and uncontrolled release of radioactive material, which could produce direct or delayed health effects on facility workers and the population that could be affected by radioactive material released.

12.2 Need for emergency planning as the last barrier of defence and mitigation of the radiological consequences of potential accidents

The primary means of preventing and mitigating the consequences of accidents is the 'defence in depth' concept (IAEA, 1996a). Defence in depth is implemented primarily through the combination of a number of consecutive and independent levels of protection that would have to fail before harmful effects could be caused to people and to the environment. Defence in depth is provided by an appropriate combination of an effective management system with a strong management commitment to safety and a strong safety culture; adequate site selection and the incorporation of good design and engineering features providing safety margins, diversity and redundancy; and comprehensive operational procedures and practices as well as accident management procedures.

Accordingly to the defence in depth concept, the design, construction and operation of nuclear facilities are conducted under the most stringent quality controls to comply with safety principles, including the development of the necessary provisions to deal with emergency situations in all modes of operation. The owner and the national authorities in charge of nuclear safety perform independent verification programmes to ensure strict compliance with safety and quality requirements.

Despite the fact that the nuclear facilities and their safety systems are designed, installed, tested, operated and verified in accordance with the strictest safety and quality standards, the possibility of an accident, a human error or an intentional action that can seriously damage the facility cannot be excluded, although its probability can be considered as extremely low. In very unlikely circumstances, these situations could cause simultaneous failure of operating and safety systems, which could produce radiation exposure of facility workers or uncontrolled discharge of radioactive material to the environment. Furthermore, in very extreme circumstances, some external phenomena, e.g. earthquake, tsunami or sabotage, could severely damage the plant, its external and internal supplies of electric power or cooling water in such a way that the operator is unable to control the safety systems. This is the situation that occurred in the Fukushima nuclear power plant on 11 March 2011 as a consequence of a big earthquake and a tsunami that partially destroyed the facility.

Both circumstances could result in damage to the health of individuals living or working near the facility as well as to their property and to the environment.

12.2.1 Emergency provisions

To mitigate potential damage that could arise from unwanted situations, every nuclear facility incorporates in its organization a number of emergency provisions, which are laid down on the assumption that the safety systems are not sufficient to control such extraordinary situations. Strictly speaking, the emergency arrangements are not part of the design of the facility, although they provide an additional guarantee of protection from the risk associated with nuclear facilities.

These arrangements should be set out in a specific emergency plan that reflects the likelihood and the possible consequences of nuclear accidents, the characteristics of the radiation risks and the nature and location of the facilities. These emergency plans are designed and prepared to ensure that arrangements are in place for a timely, managed, controlled, coordinated and effective response at the scene, and at the local, regional, national and international level, to any nuclear or radiological emergency. The emergency plans are usually structured in two levels: on-site emergency plans and off-site emergency plans.

The *on-site emergency plans* are direct targets to lead the facility that has suffered an accident to safe conditions as soon as possible, to minimize potential consequences of the accident on the staff and the installation, and to reduce the release of radioactive material to the environment. The on-site emergency plan is a primary responsibility of the plant operator and is a part of the safety documentation needed for obtaining the operating licence of every facility. The plant operator is also responsible for maintaining on-site emergency plans in an operational state by checking their effectiveness before the facility becomes operational and whilst the facility is in operation by performing suitable emergency exercises.

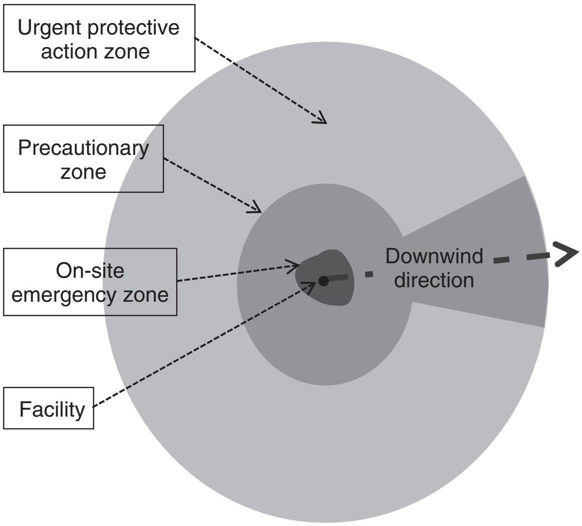
The *off-site emergency plans* are aimed at the preparation and, where appropriate, implementation of the emergency measures necessary to protect the population living around a nuclear facility against any damage caused by any accident occurring at the facility. The public authorities are responsible for designing and implementing the off-site emergency plans as a part of the national response plans established for protecting the population from any unwanted situation that could damage their health, their property or the environment. Off-site plans have to be established before every facility becomes operational and their effectiveness should be checked during the pre-operation testing period and periodically whilst the facility is in operation. Off-site emergency plans include suitable international interfaces when trans-boundary consequences of accidents that could occur in a facility are possible.

12.2.2 Emergency planning zones

Arrangements for emergency plans are carried out in several emergency planning zones that are roughly circular around the facility. The *on-site emergency zone* is the area surrounding the facility within the security perimeter. It is the area under the immediate control of the facility or operator. The *off-site emergency zone* is the area beyond that under the control of the facility operator in which intervention could be needed for emergencies resulting in major off-site releases or exposures. The level of planning will vary depending on the distance from the facility.

The off-site emergency zone is usually divided in two subzones. The *precautionary action zone* is a pre-designated area around the facility where urgent protective action has been pre-planned and will be implemented immediately upon declaration of a general emergency, to substantially reduce the risk of severe deterministic health effects by taking protective action within this zone before or shortly after a release. The *urgent protective action planning zone* is a pre-designated area around the facility where preparations are made to promptly implement urgent protective action based on environmental monitoring data and assessment of facility conditions, the goal being to avert doses specified in international standards.

Figure 12.1 shows a conceptual distribution of the emergency planning zones around a nuclear power plant.



12.1 Emergency planning zones.

12.2.3 Classes of emergency and implementation of emergency plans

According to the IAEA recommendations, the nuclear emergency plans usually consider several classes of emergency, which depend on the expected consequences of the accident scenario considered for planning:

- *General emergencies* involve an actual or substantial risk of release of radioactive material or radiation exposure that warrants taking urgent protective action off the site. Upon declaration of this class of emergency, action shall be promptly taken to mitigate the consequences of the event and to protect people within on-site and off-site zones.
- *Site area emergencies* involve a major decrease in the level of protection for those on the site and near the facility. Upon declaration of this class of emergency, action shall be promptly taken to mitigate the consequences of the event, to protect people on the site and to make preparations to take protective action off the site if this becomes necessary.
- *Facility emergencies* involve a major decrease in the level of protection for people on the site. Upon declaration of this class of emergency, action shall be promptly taken to mitigate the consequences of the event and to protect people on the site. Emergencies in this class can never give rise to an off-site area or general emergency.
- *Alerts* involve an uncertain or significant decrease in the level of protection for the public or for people on the site. Upon declaration of this class of emergency, action shall be promptly taken to assess and mitigate

the consequences of the event and to increase the readiness of the on-site and off-site response organizations as appropriate. Alerts include events that could evolve into facility, site area or general emergencies.

Design and implementation of the emergency plans are carried out in three phases:

- The *planning phase* consists of a detailed analysis of the situation that can occur at the facility. The result of this analysis is used to define the characteristics of appropriate emergency measures, to mitigate the consequences of every credible event. The results of the analysis are used to establish the emergency plan that accurately describes the organization in charge of implementing countermeasures; emergency actions to be taken in each case; a clear allocation of responsibilities of every individual participating in the implementation of emergency measures; intervention criteria; decision-making procedures for countermeasures implementation; definition of planning areas; and the means and resources needed for intervention.
- The *preparedness phase* consists of the identification, acquisition and putting into optimum use conditions of means and resources to intervene in case of emergency. Preparation includes also training of intervention personnel and maintenance of the technical, human and organizational means and resources, as well as the verification that all of them are permanently in a position to be activated. A crucial element during the preparedness phase is conducting partial or full-scale exercises for training the intervention personnel and verifying the appropriateness of emergency plans.
- The *response phase* consists of the activation of the emergency organization, as soon as possible, to cope with a real or simulated accident, through the implementation of countermeasures foreseen in the emergency plans with the available means and resources. The response phase starts with the decision of activating the plan after an accident occurs. This decision is taken by the operator in the case of on-site emergency plans and by the relevant authority in the case of off-site emergency plans. The response phase includes implementation of urgent mitigation and protective emergency measures, and undertaking appropriate remedial actions in the medium and long terms until bringing the situation to a safe condition. The response finishes when the normal situation has been recovered as far as possible.

Operators and public authorities responsible for the implementation of the nuclear emergency plans periodically conduct a review in order to ensure that situations that could necessitate an emergency intervention are identified, and to ensure that an assessment of the threat is conducted for such practices or situations. This review is undertaken periodically to take into account any changes to the threats within the State and beyond its borders, and to learn from experience and lessons from research, operating experience and emergency exercises.

12.3 International conventions and standards on emergency planning

In most countries, the nuclear emergency plans are governed by specific regulations that establish the national regimen of civil protection and radiation safety. These regulations allocate responsibilities to different actors taking part in the emergency management, set up requirements for emergency preparedness and response, and establish criteria for intervention in case of emergency.

The national civil protection legislation, which is mainly addressed to the off-site emergency plans, usually establishes the basis for planning, emphasizing the right of citizens to their own protection and their obligations in the event of emergency, as well as allocating the responsibilities of all organizations participating in the preparedness and response to nuclear emergencies. Civil protection legislation is strongly conditioned by national political and administrative structures since it sets up rights and obligations for citizens and public and private organizations, as well as the basic responsibilities and procedures to take decisions.

The national regulations on radiation safety are usually based on the standards and recommendations issued at international level for the safe and secure use of nuclear energy and its applications. Emergency preparedness and response have been taken into consideration by international standards and recommendations on radiation safety and nuclear liability, since the very beginning of the use of nuclear power for peaceful purposes. But it was in the late 1980s, as a result of the lessons learned from the Chernobyl accident, when this subject was treated by the international community as a common concern at the highest level of internationally legally binding instruments. It is too early to conclude lessons learnt from the Fukushima event, but there is no doubt that this accident will be the starting point to review some safety criteria related to facility siting and to reconsider some hypotheses usually accepted for on-site emergency plans. It is also probable that off-site emergency plans are revised to take into account the special difficulties that are expected to implement countermeasures when a big nuclear accident occurs simultaneously with a natural or anthropogenic disaster.

12.3.1 International conventions

The Convention on Early Notification of a Nuclear Accident and the Convention on Assistance in the Case of a Nuclear Accident or Radiological Emergency (IAEA, 1986) were adopted in 1986, as the most relevant legally binding instruments to establish a common framework of cooperation at international level on preparedness and response to nuclear and radiological emergencies. In parallel, some international and regional organizations, having responsibility for the use of nuclear energy, intensified their efforts for developing specific standards and recommendations to help their member states in improving and harmonizing national practices and regulations. Relevant examples of these initiatives are the publication of recommendations on interventions in the case of nuclear or radiological accidents by the ICRP; the regulation of trans-boundary movement of foodstuffs after a nuclear accident issued by the European Union (EU, 1987b) and the research projects on nuclear emergency promoted by the European Union Research Framework Programmes; the creation of the Working Party on Nuclear Emergency Matters by the OECD Nuclear Energy Agency; and the impulse given by the International Atomic Energy Agency, IAEA, to the international standards and recommendations on nuclear emergency matters.

The Convention on Early Notification of a Nuclear Accident applies in the event of any accident involving nuclear facilities or activities of a state party, or of persons or legal entities under its jurisdiction or control, in which a release of radioactive material has occurred that could be of radiological safety significance for another state. The state parties of this Convention are committed to forthwith notifying, directly or through the IAEA, those states which are or may be physically affected by a nuclear accident occurring in their territory. Every state party to this Convention is also committed to notifying the nature, the time and the exact location of the accident, as well as to providing, as soon as possible, available information to minimize the trans-boundary radiological consequences of the accident.

The state parties to the Convention on Assistance in the Case of a Nuclear Accident or Radiological Emergency are committed to cooperating between them and with the IAEA, to facilitate prompt assistance in the event of a nuclear accident or radiological emergency to minimize its consequences and to protect life, property and the environment from the effects of radioactive releases. According to this Convention, to facilitate such cooperation, state parties may agree on bilateral or multilateral arrangements for preventing or minimizing injury and damage which may result in the event of a nuclear accident or radiological emergency. The state parties can request the IAEA to use its best endeavours to promote, facilitate and support the cooperation between state parties.

12.3.2 The IAEA standards on emergency planning

Under the Convention on Assistance in the Case of a Nuclear Accident or Radiological Emergency, the IAEA has the function of collecting and disseminating to state parties and member states information concerning methodologies, techniques and available results of research relating to response to such emergencies. One of the actions undertaken by the IAEA to fulfilling its functions has been issuing several safety standards and recommendations on preparedness and response to nuclear and radiological emergencies, as a significant part of its Safety Standards Series.

The Safety Standards Series are issued by the IAEA, hereafter the Agency, in compliance with the terms of Article III of its Statute. This statutory provision authorizes the Agency, in cooperation with other relevant international organizations, to establish standards of safety for protection against ionizing radiation and to provide for the application of these standards to peaceful nuclear activities. The Safety Standards Series is composed of all regulatory related publications issued by the Agency, which covers nuclear safety, radiation safety, transport safety and waste safety, and also general safety that is of relevance in two or more of the four areas. These standards are not legally binding on Member States but may be adopted by them, at their own discretion, for use in national regulations in respect of their own activities. However, they are binding on the Agency in relation to its own operations and on states in relation to operations assisted by the Agency.

The Safety Standards Series is a set of publications structured in three levels:

- *Safety Fundamentals* set up basic objectives, concepts and principles of safety and protection in the development and application of nuclear energy for peaceful purposes.
- *Safety Requirements* establish the requirements that must be met to ensure safety. These requirements, which are expressed as 'shall' statements, are governed by the objectives and principles presented in the Safety Fundamentals.
- *Safety Guides* recommend actions, conditions or procedures for meeting safety requirements. Recommendations in Safety Guides are expressed as 'should' statements, with the implication that it is necessary to take the measures recommended or equivalent alternative measures to comply with the requirements.

Many publications in the Safety Standards Series include rules and recommendations applicable to the nuclear emergency preparedness and response. The most recent restructuring of the series, dated 2006, considers nuclear and radiological emergencies as a general safety topic that has to be taken into account in every nuclear radiation facility or activity and it is treated at all level of the safety documents.

Principle 9 of the Fundamental Safety Principles is the basis for the standards and recommendations on nuclear emergency matters in the Safety Standards Series. Principle 9 is titled: 'Arrangements must be made for emergency preparedness and response for nuclear or radiation incidents' (IAEA, 2006, p. 14). According to this safety principle (IAEA, 2006, paragraph 3.34) the primarily goals of preparedness and response for a nuclear emergency are:

- 'To ensure that arrangements are in place for an effective response at the scene and, as appropriate, at the local, regional, national and international levels, to a nuclear or radiation emergency;
- To ensure that, for reasonably foreseeable incidents, radiation risks would be minor;
- For any incidents that do occur, to take practical measures to mitigate any consequences for human life and health and the environment.'

The scope and extent of arrangements for emergency preparedness and response have to reflect the likelihood and the possible consequences of a nuclear or radiation emergency, the characteristics of the radiation risks, and the nature and location of the facilities and activities.

The recommendations of the ICRP on nuclear and radiation emergency matters (ICRP, 1991) are the main basis of the Agency radiation safety standards, and its principles and recommendations are endorsed by the Agency in the document *International Basic Safety Standards for Protection against Ionizing Radiation and for the Safety of Radiation Sources* (IAEA, 1996b) that is undergoing a deep revision, now in an advanced stage of development, which will be issued as a Generic Safety Requirement (GRS Part 3). The Basic Safety Standards are also based on assessments of the biological effects of irradiation made by the United Nations Scientific Committee on the Effects of Atomic Radiation, UNSCEAR, and on the recommendations of the Agency's International Nuclear Safety Group, INSAG.

The Basic Safety Standards were co-sponsored by the Food and Agriculture Organization of the United Nations, FAO, the International Labour Organization, ILO, the Nuclear Energy Agency, NEA, the Pan-American Health Organization, PAHO, and the World Health Organization, WHO, and represent an international consensus on qualitative and quantitative requirements for protection and safety for planned practices such as nuclear power generation and also for intervention in existing situations such as exposure following an accident. It is the most relevant international reference to establish national and regional regulation on radiological protection on, among other relevant topics, occupational radiation protection,

protection of the public and the environment from exposure to radioactive materials released to the environment, prevention of incidents giving rise to potential exposures, and intervention in a radiological emergency.

The Basic Safety Standards set out the basic requirements for nuclear and radiological emergency management, providing radiological criteria applicable to emergency response. They are established under the belief that, in most cases after a nuclear or radiological accident, emergency actions are needed if dose rates generated by the accident or the doses that can be prevented by applying emergency measures can lead to significant radiation injury. The Basic Safety Standards also provide guidelines for the implementation of the optimization principle to the measures to be applied in an emergency.

According to the Basic Safety Standards, the protective actions implemented to respond to an emergency situation should be oriented to protect individuals potentially affected by the accident, which includes the emergency workers and members of the public. Implementation of protective actions should be based on intervention levels expressed in terms of doses that can be avoided with the intervention, considering different exposure pathways, including direct irradiation by radiation emission coming from radioactive contamination of air, soil or water. The decision to implement any countermeasure should be based on the circumstances that actually exist when an emergency happens and, if possible, be taken in anticipation of a possible radioactive release better than when the issue has been confirmed. The main protective actions recommended to protect individuals are sheltering and evacuation or prophylaxis with stable iodine to prevent internal contamination with radioactive iodine, of people potentially affected by an accidental release. In some special cases, decontamination of individuals and goods could be recommended to reduce the dose and spread of contamination to non-affected areas.

The Basic Safety Standards also give recommendations to prevent chronic exposure by controlling the use of contaminated land and facilities and the consumption of contaminated food and water. Reference levels set up by the FAO-WHO Codex Alimentarius Commission (FAO-WHO, 1991) are recommended to adopt such decisions. In some cases, these countermeasures should be implemented after a careful optimization process taking into account the averted dose and the social and economic consequences of implementing such measures. In this regard, reasonable steps have to be taken to assess exposure incurred by members of the public as a consequence of a nuclear accident, and the results of this assessment should be made public and periodically updated to optimize the implementation of protective measures, until conditions allow the implementation of protective actions according to intervention levels to be discontinued.

In 2002, the Agency issued a Safety Requirements document (IAEA, 2002) specifically applicable to preparedness and response for a nuclear or radiological emergency, which incorporates and establishes requirements so that emergency management can be seen in its entirety by the bodies concerned. This document was co-sponsored by the above-mentioned international organizations and the United Nations Office for the Co-ordination of Humanitarian Affairs, which were concerned with the harmful potential consequences of nuclear accidents, as a result of the large impact of the Chernobyl accident.

These Safety Requirements compile, organize and augment all the requirements relating to emergency management established in other publications of the Agency. The Safety Requirements are applicable to all those nuclear and radiation practices and sources that have the potential for causing radiation exposure or environmental radioactive contamination warranting an emergency intervention, particularly to facilities hosting nuclear reactors and other nuclear fuel cycle facilities.

The Safety Requirements establish requirements for an adequate level of preparedness and response for a nuclear or radiological emergency in any State. Their implementation is intended to minimize the consequences for people, property and the environment of any nuclear or radiological emergency. The fulfilment of these requirements also contributes to the harmonization of arrangements in the event of a transnational emergency. These requirements are intended to be applied by authorities at the national level by means of adopting legislation, establishing regulations and assigning responsibilities, and also apply to the off-site jurisdictions that may need to make an emergency intervention in a State.

The Agency also provides recommendations for the implementation of these Safety Requirements in a Safety Guide (IAEA, 2007), which is intended to assist Member States in the application of the Safety Requirements on Preparedness and Response for a Nuclear or Radiological Emergency, and to help in fulfilling the Agency obligations under the Assistance Convention. The Safety Guide provides basic concepts that must be understood to apply the guidance and to meet the requirements, discusses the concept of operations, and describes in general terms how the response should proceed for different types of emergency.

These standards and recommendations are complemented with a series of technical documents, called Emergency Preparedness and Response, oriented to assist Member States in the development of their own capacities to respond to nuclear and radiological emergencies (IAEA, 2003a). This series provides recommendations to establish suitable methods for, among other relevant topics, developing arrangements for response to a nuclear accident; organization and training of first responders to a nuclear emergency; preparing, conducting and evaluating emergency exercises; providing medical assistance in case of nuclear and radiation accidents; using radiometric instrumentation for the response to nuclear and radiation emergencies; and developing and implementing adequate procedures for prompt notification and mutual assistance.

12.4 Responsible organizations

In the event of a nuclear emergency the time available for decision making and for implementing an effective strategy for response may be short. It is therefore important that an appropriate management system be used. All organizations that may be involved in the response to a nuclear or radiological emergency shall ensure that appropriate management arrangements are adopted to meet the timescales for response throughout the emergency. Where appropriate, the management system shall be consistent with that used by other response organizations in order to ensure a timely, effective and coordinated response.

Although its probability is extremely low, the possibility that a large-scale nuclear accident has adverse health, social, economic, political and transnational consequences implies that national authorities have a key role in the development and implementation of nuclear emergency plans. Jurisdictions of the various orders and levels of government may be laid out in substantially different ways between States. Likewise, the authorities of the various organizations that could be involved in emergency response may be allocated in substantially different ways. A generic approach to describe the allocation of responsibilities at national level can be gotten from international recommendations. The national legislation allocates clearly the responsibilities establishing or identifying an existing governmental body or organization to act as a national coordinating authority. This authority is in charge of ensuring that responsibilities of operators and response organizations are clearly assigned and are understood by all response organizations, and that arrangements are in place for achieving and enforcing compliance with the requirements.

The plant operator, the employer, the regulatory body and appropriate branches of government are responsible for establishing arrangements for preparedness and response for a nuclear or radiation emergency at the scene, at local, regional and national levels and, where so agreed between States, at the international level. In practice, all levels of national, regional and local public authorities have some responsibility in preparedness and response to nuclear emergencies. Responsibilities assumed by each authority depend on national political and administrative organization. Notwithstanding, sharing of responsibility is usually established according to a common pattern.

12.4.1 Responsibilities of the licensee

Primary responsibility in responding to a nuclear emergency falls on the operator and deals with implementation of provisions of on-site emergency plans under the oversight of the regulatory authority, as part of the requirements established in the nuclear safety regulations. In discharging this responsibility, the operator has the following functions:

- Designing, building and operating nuclear plant in such a way that the probability of a breakdown, an accident or a malicious act that could trigger an emergency is minimized
- Developing on-site emergency plans appropriate for its facility and site, providing them with the necessary resources, and keeping them fully operational
- Providing to its staff special training on crisis management and operation of the plant under emergency conditions
- Conducting periodic emergency exercises and drills to train its staff and to check the full operability of its on-site emergency plan
- Having suitable procedures and resources to bring the plant to safe conditions in the shortest time possible, and to implement them as soon as possible to minimize the health risks to its own staff and the uncontrolled release of radioactive material abroad, in case of an emergency situation
- Cooperating with authorities in the preparation of emergency response by providing the means, resources and information necessary to draw up plans for protecting the population
- Notifying urgently to authorities in charge of protecting the population, the occurrence of any situation that requires the activation of contingency plans designed to protect people and keep them informed of developments, providing all available information that can be useful to take decisions and optimizing the use of the resources available to emergency plans. In many countries, licensees are responsible for taking first-response decisions until public authorities have been activated and are ready to assume the direction of the emergency response.

12.4.2 Responsibilities of the national authorities

National authorities assume overall responsibility for implementing adequate nuclear emergency plans to protect people, the economy and the environment from radiation hazard in case of nuclear accidents. In this regard, they are responsible for adopting social, political, economic and technical measures dealing with major nuclear emergencies. In discharging these responsibilities, national authorities usually assume the following basic and strategic functions:

- Issuing appropriate legislation and regulations for nuclear emergency planning, preparedness and response
- Approving nuclear emergency plans ensuring clear allocation of responsibilities among organizations involved and establishing appropriate procedures for making decisions and their implementation, consistent with other national emergency response plans
- Providing funds, means and resources necessary to ensure effective implementation of nuclear emergency plans; provisions have to include extraordinary resources to respond to large-scale nuclear emergencies derived from extreme severity accidents, including adequate facilities for medical treatment of large numbers of victims
- Establishing training strategies and programmes applicable to all levels of responsibility within the organizations that have been assigned a role in nuclear emergency plans
- Activating nuclear emergency plans as necessary, as well as directing the response operations until people and areas affected by the accident have recovered normal conditions as far as possible
- Adopting necessary measures to limit long-term radiological consequences resulting from contamination caused by nuclear accidents, such as control of trade and consumption of contaminated products; limitation of the use of land, water, areas, facilities and property affected by a nuclear accident; and implementation of hard countermeasures to revert to normal conditions
- Coordination of all activities related to public information concerning the preparation of emergency plans and response in case of emergency, paying special attention to putting into practice specific information programmes to ensure that people who could be affected by a nuclear emergency are aware of their potential risk and know how to behave in an emergency
- Establishing adequate mechanisms to ensure monetary compensation derived from civil liability of parties involved in the origin and following up of a nuclear emergency
- Establishing adequate arrangements for prompt notification to international organizations and the authorities of other countries potentially affected by a nuclear accident occurring in its own territory, as well as for providing or receiving assistance in nuclear emergencies.

12.4.3 Responsibilities of the regulatory body

The main responsibility of the regulatory body regarding emergency planning and preparedness is to ensure that emergency arrangements are integrated with those of other response organizations as appropriate before the commencement of operation. The regulatory body ensures that such emergency arrangements provide a reasonable assurance of an effective response, in compliance with safety requirements, in the case of a nuclear or radiological emergency. In discharging this responsibility, the regulatory body assumes the following regulatory functions:

- Establishing radiological criteria for emergency planning, which include, among others, definition of intervention zones according to dose rate or surface contamination levels, adequate countermeasures to protect personnel, the public and the environment, and quantitative reference levels to undertake countermeasures
- Issuing regulation and acceptance criteria for on-site emergency plans, and giving guidance to licensees to develop and implement on-site emergency plans
- Evaluating and approving on-site emergency plans drawn up by the owners as part of the safety documentation required for applying for the authorization of each facility
- Verifying that on-site emergency plans are established according to the applicable regulation, by auditing and inspecting them, and supervising the conduct of pre-operational and periodic emergency exercises
- Requiring modification of emergency plans if it considers that they are inadequate for the facility and site characteristics, the state-of-the-art recommends improving them, or when a new regulation has entered in force
- Advising, supervising and, when needed, requiring implementation of emergency countermeasures to ensure that the exposure of intervention personnel and other affected persons is kept as low as possible and to ensure that actions undertaken to return to normality are carried out in accordance with radiation safety regulations
- Advising national authorities to fulfil their international commitments arising from multilateral or bilateral agreements signed by the State in the field of nuclear emergency.

12.4.4 Responsibilities of regional and local authorities

The preparedness for nuclear emergency requires many arrangements in emergency intervention zones established in the surroundings of nuclear facilities according to regulatory criteria. Regional and local authorities play an important role in the implementation of such arrangements, since they have a detailed knowledge of the geographical, economic and social conditions of these zones. The responsibilities and functions of regional and local authorities are usually targeted to address logistic and operational issues of the nuclear emergency plans. In discharging these responsibilities, regional and local authorities usually assume the following logistic and operational functions:

- Establishing local action plans and procedures to ensure that emergency countermeasures can be implemented in such a way that every potentially affected person will be adequately protected in case of emergency
- Providing adequate facilities to implement emergency countermeasures, including adequate centres to concentrate, monitor, decontaminate and take social care of victims, including relocation if needed
- Providing medical facilities and resources adequately equipped for first aid involving medical care of victims potentially irradiated or contaminated
- Providing adequate facilities to store emergency equipment in suitable conditions to be used in case of emergency
- Establishing and implementing training programmes for intervention personnel, and promoting their participation at all levels of responsibility
- Preparing, conducting and evaluating periodic drills and exercises organized to train responders and verify plan effectiveness
- Develop and put in practice adequate public information programmes aimed at teaching people how to protect themselves in case of emergency, and efficiently transmit information needed to manage the emergency in the most efficient way.

12.4.5 Responsibilities of specialized organizations

A nuclear emergency could lead to very complex situations that require the intervention of a number of specialized organizations to implement adequate countermeasures.

- Fire and rescue brigades, which are responsible for providing help to make the plant safe and to assist the victims most severely affected by conditions derived from and during the emergency.
- Police organizations, which are responsible for maintaining order if controlling access to the affected areas or evacuation of affected people is needed.
- Medical services, which are responsible for providing medical assistance to victims affected by radiation, contamination or as a consequence of the implementation of countermeasures. Medical services are also responsible for providing specific health care by using prophylaxis with stable iodine, which could be a very effective countermeasure to prevent radiation injuries produced by inhalation of radioiodine that is released in case of severe damage to a nuclear reactor, because of its volatility.
- Social services, which are responsible for providing assistance to victims and providing them with adequate first-aid and relocation settlements when necessary. Psychological care of victims of a nuclear accident could

also be an important task of social services because the population is not familiar with radiation risk and this can produce anxiety in some cases.

The radioactive material released in the case of a large nuclear emergency comprises mainly gases or aerosols. Liquid releases are easily isolated and it is unlikely that a huge amount of radioactive materials would escape in liquid form. In some cases, gaseous releases could be impelled by the large amount of energy accumulated or produced in the nuclear facility, particularly if the facility is a nuclear reactor. This energy can contribute to spreading the releases into high levels of the atmosphere and reaching long distances. In these circumstances the information provided by meteorological services is crucial in predicting the scope of the contamination and the areas that can be affected as a consequence of the emergency.

In addition, the response to very large nuclear emergencies affecting large geographical areas could require also the participation of many different organizations specialized in topics such as facilities decontamination, radiation environmental surveillance, radio-epidemiology and radio-ecology. Usually these organizations are national institutions which have many international interfaces with homologous organizations from other countries or international organizations. The experience gained from the long-term response given to the Chernobyl accident shows that this kind of international cooperation is a proper way to share knowledge and optimize resources in responding to nuclear emergencies.

12.5 Emergency management

According to the international recommendations (IAEA, 2007) on nuclear emergency matters, the practical goals of emergency response are:

- To regain control of the situation
- To prevent or mitigate consequences at the scene
- To prevent the occurrence of deterministic health effects in workers and the public
- To render first aid and to manage the treatment of radiation injuries
- To prevent, to the extent practicable, the occurrence of stochastic health effects in the population
- To prevent, to the extent practicable, the occurrence of non-radiological effects on individuals and among the population
- To protect, to the extent practicable, property and the environment
- To prepare, to the extent practicable, for the resumption of normal social and economic activity.

The goals of emergency response are most likely to be achieved in accordance with the principles for intervention by having a sound programme for emergency preparedness in place as part of the infrastructure for protection and safety. The practical goal of emergency preparedness is to ensure that arrangements are in place for a timely, managed, controlled, coordinated and effective response at the scene and at local, regional, national and international levels to any nuclear or radiological emergency.

Nuclear emergency preparedness is a long and continuous process that begins with the selection of the site to build a nuclear facility, giving due consideration to the circumstances – geographical, demographic, geological, hydrological, agricultural and social – that characterize the selected location, continues during the design and construction phase with the implementation of emergency systems and procedures, and is completed while the plant is in operation through the maintenance plan operability.

12.5.1 The design phase

In designing a nuclear facility a comprehensive safety analysis is carried out to identify all sources of exposure and to evaluate radiation doses that could be received by workers and the public, as well as the potential impact the facility can have on the environment. Every event sequence, including those originated by extreme external phenomena, that may lead to an accident is examined in the safety analysis, and the results are used as the basis for designing emergency arrangements, which include:

- Designing and ensuring full operability of the safety system installed to mitigate accidental sequences leading to uncontrolled releases of radioactive material to the environment or producing damage to the plant and the unwanted exposure of its personnel
- Implementing adequate operational procedures to lead the plant to safe conditions after any accident or malicious acts that can seriously damage reactivity control systems, cooling systems or confinement of radioactive material systems
- Implementing a training programme to ensure that all personnel have adequate skills to manage any crisis generated by any situation that can lead to an emergency situation.

During plant operation emergency preparedness activities are focused on maintaining and improving the capacity to manage any emergency situation. These include:

- Ensuring the operability of systems installed to control any credible accidental situation identified in the risk assessment, including adequate security arrangements to deter, detect and respond to malicious acts
- Implementing adequate training programmes to maintain the emergency management skills of the plant's personnel at the maximum level

- Conducting an adequate drill and exercise programme to validate, train and enhance the emergency plans and procedures.

12.5.2 The initiation phase

A nuclear emergency starts when the plant monitors indicate that some operational systems do not operate properly and the situation cannot be controlled by the corresponding safety systems adequately. Upon failure detection, the plant operator evaluates the impact of the incident on the plant and identifies the affected systems and the availability of alternative systems to control the situation. Simultaneously, the plant operator investigates whether the incident could lead to the escape of radioactive material within the plant or to the environment. Based on the results of its preliminary evaluation, the operator initiates the mitigation actions, decides the level of the on-site emergency plan to be activated, and notifies the situation to the emergency coordinator who is responsible for off-site emergency plan activation.

Transition from normal operation to an emergency situation is a critical step that needs to be clearly established in emergency plans; for this reason, the operators are specifically trained in the use of procedures to identify abnormal situations within the plant, activate on-site emergency plans, and implement the emergency procedures to handle the situation. Similarly, activation of the off-site emergency plans requires specific training of the authorities in charge of response and intervention. The on-site/off-site interface also needs careful implementation to avoid any delay or disturbance in taking the necessary countermeasures.

12.5.3 The active phase

Once emergency plans have been activated, the operator is responsible for timely and accurate transmission of information about the evolution of the accident within the plant, to ensure that the public authorities receive data they need for managing the situation. Of special importance are data related to the nature and amount of the radioactive releases from the plant, usually called the source term, because the scope and nature of countermeasures to be implemented depend critically on this parameter. The source term can be evaluated by using data from of the radiometers installed in the main discharge channels in the nuclear facility, e.g. chimneys and ventilation exhaust systems. This method can be used when radioactive materials are released through these channels and the corresponding instruments were not affected by the accident. The source term can also be estimated by using mathematical models that reproduce the physical–chemical behaviour of the plant under accident conditions. Some of these have been adapted for use in emergency situations and are available in the emergency coordination centres operated by operators and regulatory authorities (IAEA, 2003b).

In case of maximum severity, the emergency coordinator can decide the implementation of precautionary urgent protective action to prevent severe deterministic health effects by keeping doses below those for which intervention would be expected to be undertaken under any circumstances. This situation is extremely unlikely and it is expected that the emergency coordinator would have enough time to decide the implementation of countermeasures based on dose estimation.

Estimation of the dose needs detailed meteorological data that can be obtained from the stations existing in every nuclear facility and from regional or national meteorological services. These data are used as input to mathematical models able to predict transport of radioactive materials released in the atmosphere, and to estimate the dose that the people living in the areas affected could receive due to radiation from a contaminated cloud or radioactive aerosols deposited in soil or waters. The dose can also be evaluated by using the radiometric instrumentation that is available in the emergency areas as part of the means and resources arranged during the preparedness phase of the emergency plans. This instrumentation is composed of automatic radiation surveillance networks, mobile units, personal dosimeters, contamination meters, and sampling stations and analytical laboratories and procedures to evaluate contamination of affected pathways, e.g. air, soil, foods and water.

The emergency coordinator can evaluate the radiological situation by using the different methods available to estimate the source term, the spread of contamination and the dose. Use of an adequate technique is a compromise decision between the need for quick or accurate results. The use of mathematical models allows very quick results, even predictive, but can involve some uncertainties. The use of radiometric measures is more accurate but can lead to delay, especially if the results are obtained by sample analysis or with off-line instruments. Automatic radiation surveillance networks can reduce the time needed to obtain results, but their accuracy, sensitivity or location could be inadequate for taking decisions. The emergency coordination centres are equipped with systems based on different techniques and their operators are trained in using all of them and taking decisions based on combining the results obtained from all of them.

Upon consulting with the regulatory authority, the emergency coordinator decides the implementation of urgent protective action to prevent stochastic effects to the extent practicable by averting doses, in accordance with international standards. The decision is based on the dose rate and contamination levels existing in the affected area, and the dose that can be averted by applying appropriate countermeasures. Public authorities can take into consideration other factors influencing the implementation of countermeasures. In this regard, meteorological conditions, seasonal demography and coincidence with other catastrophic events such as earthquakes are examples of circumstances that have to be taken into account in the decision. Finally, the emergency coordinator transmits his or her decisions about emergency actions to emergency response teams (rescue brigades, radiation protection, health services, police, and civil defence teams) for implementation.

During emergency response, the emergency coordinator has to pay special attention to ensure that easily understandable information about existing hazards, emergency decisions and countermeasures to be implemented is properly transmitted:

- Directly to the permanent, transient and special population groups or those responsible for them and to special facilities within the emergency zones, for getting an adequate undertaking of emergency decisions and their full collaboration in implementing emergency measures
- To emergency coordination centres to act cooperatively
- To the media to ensure that all stakeholders have adequate information on emergency operations to act properly if their support is required
- To international partners, international organizations and national signatories of bilateral agreements, to facilitate the adoption of adequate emergency actions in their own territories by the relevant emergency coordinator.

During emergency operations, special attention should be paid to protect emergency responders who may undertake intervention in order to save lives or prevent serious injury due to doses that could cause severe deterministic health effects, take action to avert a large collective dose, or take action to prevent the development of catastrophic conditions.

Activation and implementation of emergency plans could be especially difficult when response to a nuclear accident has to be given in an area that has been simultaneously affected by an extreme natural or anthropogenic disaster. In this case, the emergency coordinator has to pay special attention to coordinating implementation of radiological and non-radiological countermeasures with the relevant authorities.

12.5.4 The post-emergency phase

When a nuclear accident involves a large amount of radioactive aerosols, the subsequent fallout may contaminate the soil and water of affected areas. In this case, public authorities should implement specific protective measures, to protect the public in accordance with international standards, from contamination through foodstuff and water consumption and inhalation of airborne aerosols deposited in soil. It is likely that these actions would have to be continued for a long period. Management of these situations is usually beyond the scope of the nuclear emergency plans, because they could involve the intervention of a wide range of government institutions and require special political and economic decisions.

Lessons learned from the aftermath of the Chernobyl accident show that a relevant fraction of the total dose produced by a large nuclear accident arises from chronic exposure. Chronic exposure is mainly produced by direct exposure to soil contamination, contaminated food consumption and inhalation of radioactive airborne materials. Reducing long-term exposure could require implementation of hard countermeasures such as the long-term health care of victims, modification of agricultural practices and strict control of foodstuffs, and relocation of the population living in affected areas. Putting these countermeasures into practice requires spending large amounts of resources and can produce significant social and psychological effects on the affected population. Recent studies carried out in the region affected by the Chernobyl accident show that training the population to live with enhanced levels of radioactivity can help them to reduce the social and physiological effects and contribute decisively to normalizing the situation. The aftermath of the Chernobyl accident shows also that the efficient implementation of adequate countermeasures in the early emergency phase can reduce some long-term health effects such as thyroid cancer produced by inhalation and ingestion through milk of radioactive iodine.

12.6 Emergency drills and exercises

The adequacy of emergency response arrangements can be evaluated through the audit and review of plans, procedures and infrastructure (preparedness). The ability to carry out the required emergency actions (response) can be assessed through audits and reviews of past performance, but it is most commonly evaluated through exercises.

Emergency response exercises are a key component of a good emergency preparedness programme. They can provide a unique insight into the state of preparedness of emergency response organizations. They can also be the basis for continued improvement programmes for the emergency response infrastructure. However, to be most useful, emergency response exercises need to be well organized, professionally conducted and focused on the potential for constructive improvement. Nuclear emergency response exercises are a powerful tool for verifying and improving the quality of emergency response arrangements. Each exercise represents a significant investment of effort, financial resources and people. It is therefore important for each exercise to yield the maximum benefit. That benefit depends primarily on the quality of the preparation, conduct and evaluation of the exercise.

An emergency response exercise is not an isolated event, but rather a part of an overall exercise programme that is normally implemented over a cycle of several years. This cycle includes several types of emergency exercise. The programme is conducted to validate emergency plans and procedures and to test performance, to train intervention personnel in a realistic situation, and to explore and test new concepts and ideas for emergency arrangements. Emergency preparedness programmes should also include considerations and arrangements for international liaison, notification, exchange of information and assistance. According to the IAEA recommendations (IAEA, 2007) a cycle of emergency exercises includes several types of drills and exercises. The most common are as follows.

- *Drills* normally involve small groups of persons in a learning process designed to ensure that essential skills and knowledge are available for the accomplishment of non-routine tasks such as emergency radiation measurements or the use of emergency communication procedures. A drill can also be used to assess the adequacy of personnel training and is usually supervised and evaluated by qualified instructors. It normally covers a particular component, or a group of related components, associated with the implementation of the emergency plan and is conducted several times per year.
- *Tabletop exercises* are discussion-type workouts conducted around a table. All the participants are in the same room or building and no communication link with any outside body is necessary. They are not usually conducted in real time and their main focus is on decision-making, assessment, public and media communication policy definition, and implementation.
- *Partial and full-scale exercises* are simulations used to allow a number of groups and organizations to act and interact in a coordinated fashion. The focus of partial and full-scale exercises is on coordination and cooperation. Exercises can be partially or fully integrated. The integrated full-scale exercise involves the full participation by all on-site and off-site response organizations. Its major objective is to verify that the overall coordination, control, interaction and performance of the response organizations are effective and that they make the best use of available resources. Combined on-site/off-site exercises are usually performed to test both the on-site and off-site responses and the interface mechanisms in place, which are so important to a proper overall response. In fact, the interface aspects are often the weak link in the emergency response system and need to be tested and updated frequently. When appropriate, partial and full-scale exercises should be organized to train

intervention organizations to respond simultaneously to a nuclear accident and a natural or anthropogenic disaster affecting the same areas.

- *Field exercises* focus on the tasks and coordination of resources that must be operated at or around the site of an emergency. Those include means used by survey teams, police, medical first-aid and fire-fighting teams. Field exercises are conducted on their own or combined with a partial or full-scale exercise. Figure 12.2 shows first responders preparing to intervene in a nuclear emergency field exercise carried out by the Ministry of Interior, the Ministry of Defence and the Nuclear Safety Council in Madrid (Spain) in 2010.

The frequency of integrated exercises is a matter to be determined by the regulatory authorities. Usually an integrated exercise is conducted in every nuclear facility every year. After every emergency exercise, a performance evaluation is conducted to identify areas of emergency plans and preparedness that may need to be improved or enhanced. As a result of an exercise evaluation, there may also be recommendations on ways to correct the identified deficiencies, problems or weaknesses.

Several international organizations conduct nuclear emergency exercises at different scales. Significant examples of these international exercises are ConvEx exercises organized by the IAEA (Martincic and Obrentz, 2008), INEX exercises organized by the Nuclear Energy Agency (NEA, 2007) and ECURIE exercises organized by the European Commission (EU, 1987a).

12.2 Nuclear emergency field exercise carried out by the Ministry of Interior, the Ministry of Defence and the Nuclear Safety Council in Madrid (Spain) in 2010 (courtesy of M. Gutierrez, Ministry of Interior, Spain).

12.3 Follow-up of an international nuclear emergency exercise at the IAEA Incident and Emergency Centre (courtesy of IAEA).

Similar exercises are conducted by other international organizations at regional level in America and Eastern Asia. In addition, some international organizations, such as the North Atlantic Treaty Organization, NATO, the World Meteorological Organization, WMO and the World Health Organization, WHO, organize nuclear or radiological emergency exercises focused on topics under their specific responsibilities. Figure 12.3 show the follow-up of an international nuclear emergency exercise at the IAEA Incident and Emergency Centre.

12.7 Emergency coordination centres

The occurrence of a nuclear emergency will lead to a sequence of response actions focused on managing the incident and mitigating its effects (the responsibility of the site operator), and protecting the public against actual or potential effects of the incident (the responsibility of the site operator, and governments through the respective emergency planning and preparedness authorities). Many activities will be undertaken by the operator and respective orders of government (local, regional, national and, where appropriate, international or neighbour countries) for responding to the emergency in a timely and adequate way.

Rapid and effective coordination among all organizations involved is a crucial issue for a successful response. Coordination among these organizations requires implementation of a well-structured action plan based on an efficient network of command and control centres. Usually, every response organization has its coordination centre to command its tasks and coordinate them with the rest of the response organizations. Every command and control centre should have clearly established its role and be endowed with sufficient human and technical resources to fulfil its mission.

12.7.1 Coordination centres operated by licensees

The operator usually has two emergency centres closely interconnected. The first centre is located in the facility and the second in a location outside the areas likely to be affected by the emergency. In addition to these two emergency centres, the licensees also operate an emergency centre in the company headquarters, mainly to be informed, but also to help the reactor operators in the implementation of severe accident guidelines and procedures. The functions of the operator's centres are:

- To provide technical support to the personnel operating to bring the plant to safe conditions as soon as possible and minimize the impact of the emergency on the facility and its workers as well as reduce the uncontrolled release of radioactive material to the environment
- To identify and request external aid required for the plant, according to the evolution of the emergency
- To provide public authorities that manage the external emergency with available information on the emergency at the facility, to facilitate the implementation of measures to protect the population
- To direct urgent off-site emergency activities until the public authorities assume direction of operations.

To adequately fulfil this mission the operator's operational centres have access to all available information on plant design and operation; adequate procedures to operate the plant in degraded safety conditions; simulators of the behaviour of the plant that can predict the evolution of any event and anticipate the most appropriate mitigation measures; detailed information on the geography and on-line meteorological data for the site to evaluate atmospheric dispersion of potential uncontrolled release of radioactive material by air; redundant connections to the emergency coordination centres used by public authorities to take decisions, to inform them on the development of on-site emergency and request their help if necessary; direct connections with suppliers of equipment and services; connections with the nuclear facilities or similar technology and electricity generation of other countries; and connection with centres of water resources management, national meteorological services and other relevant coordination centres that operate networks or systems relevant for emergency response.

12.7.2 The national emergency centres

The command and control centres of the national authorities are usually set up to respond to every kind of emergency situation and therefore their technical capabilities are not specifically designed for responding to nuclear emergencies. These centres are equipped with powerful and versatile communication devices, and are directly connected with other centres to respond to national crises at the highest level. The mission of these centres is to provide the political support, strategic coordination, public information management and international contacts needed to manage the emergency. In the case of activation of a nuclear emergency plan, representatives from different ministerial organizations are convened to this centre to facilitate a response that involves several government functions such as civil protection, public security, health, environmental protection, industrial policy, finance and international relations. A key role of these centres is the coordination among public authorities at local, regional and national level, which could be a crucial element to ensuring a proper use of available resources for the implementation of emergency measures.

12.7.3 Coordination centres in regulatory organizations

The regulatory bodies usually have coordination centres that are highly specialized to respond to nuclear and radiological emergencies. The main mission of these centres in the case of a nuclear emergency is to provide national authorities with timely and accurate technical information and give recommendations for managing the emergency situation. To do that, these centres are usually designed to process information received from every nuclear power plant, from national meteorological services, from environmental surveillance networks, and from other technical sources, to assess the evolution of the consequences of the emergency situation in terms of dose rate existing or predicted in the affected areas. The result of these assessments is used to recommend emergency measures to national, regional and local authorities depending on projected dose in accordance with the evolution of the accident and the meteorological conditions around the plant and the affected areas. These centres are usually equipped with sophisticated devices and systems able to catch and transmit large amounts of technical data on the operational situation of the facility originating the emergency, the radiological situation within and outside the facility, the meteorological situation and forecast, and other technical and environmental data, and to process them and give recommendations to the national authorities concerning the implementation of emergency countermeasures.

The emergency centres of regulatory bodies are usually equipped with access to automatic radiological environmental surveillance networks. These networks cover all of the national territory and are denser in the vicinity of the nuclear facilities. They are designed to detect and give an urgent and independent warning on atmospheric releases above certain threshold levels set up as a function of intervention levels.

In many countries the emergency centres of regulatory bodies also have a very important role regarding public information in the case of a nuclear emergency, since they are able to provide accurate and independent technical information. In this regard, they are responsible for classifying the nuclear emergency in accordance with the International Nuclear Event Scale (IAEA, 2009) issued by the IAEA as a simple and common tool to communicate the severity of nuclear events all over the world. It is also very common that the emergency centre owned by the regulatory body acts as the national contact point regarding the international conventions on early notification and mutual assistance in the case of nuclear and radiological events. Figure 12.4 shows the emergency operational centre (Sala de Emergencias, Salem) of the Spanish nuclear regulatory body (Consejo de Seguridad Nuclear, CSN).

12.7.4 International coordination centres

Several international organizations operate emergency centres partially or fully devoted to responding to nuclear or radiological emergencies. The IAEA operates the Incident and Emergency Centre as the global focal point for responding to nuclear and radiological emergencies under the terms established in the Conventions on Early Notification and Mutual Assistance. The Centre provides round-the-clock assistance to Member States and coordinates the drafting and publication of the IAEA standards and recommendations on emergency matters. The Centre also organizes training activities and international nuclear emergency exercises called ConvEx aimed at verifying international cooperation in responding to nuclear emergencies. Figure 12.5 shows the Incident and Emergency Centre of the International Atomic Energy Agency.

12.4 The Emergency Operational Centre (Sala de Emergencia, SALEM) of the Spanish Nuclear Regulatory Authority (Consejo de Seguridad Nuclear, CSN) (courtesy of the CSN).

12.5 The IAEA Incident and Emergency Centre (courtesy of the IAEA).

12.7.5 Regional and local emergency centres

Regional and local authorities have their own operations centres whose mission is to implement emergency operations. These centres, which in many cases are also the centres of non-nuclear emergency management, are endowed with specific media to stay permanently and securely connected with advanced command posts that are responsible for the implementation of countermeasures, as well as with other focal points, in order to:

- Receive the information sent by the operator's emergency centre on the possible evolution of an emergency in the affected facility, and by the emergency centre of the regulatory body, giving technical recommendations necessary to implement the appropriate emergency measures to protect the population
- Send orders to every intervention team
- Transmit operational information to local media
- Inform national authorities on the evolution of an emergency in the affected area and seek their help if they need means of intervention or extraordinary resources that are not available in the territory under the operator's control.

12.8 Sources of further information and advice

Many national and international organizations related to nuclear energy, emergency management and radiation safety research have devoted efforts to issuing information on nuclear emergency matters. This information covers a wide range of topics, orientations, objectives and kinds of documents, and can be consulted on the corresponding websites. In this regard:

- The IAEA, as mentioned above, has issued a number of safety standards, recommendations and technical documents oriented at providing the Member States with adequate information for planning, preparedness and response to nuclear emergencies.
- The European Commission has contributed to the current knowledge of nuclear emergency management from a number of research projects carried out with its Framework Research Programmes during the last three decades. Significant results of these projects have been issued by the Commission and its associate research centres.
- The Nuclear Energy Agency has a Working Party on Nuclear Emergency Matters that acts as a forum for discussing development made by Member States on this subject.
- At national level, national emergency agencies and regulatory authorities permanently hold information and maintain programmes addressed to nuclear and radiological emergencies. These programmes include issuance of regulations, guidance and technical documents that are easily available directly from these institutions.

12.9 References

EU (1987a), Council Decision of 14 December 1987 on Community arrangements for the early exchange of information in the event of a radiological emergency. *EU Official Journal* L371, 30 December 1987, pp. 76–78, Brussels.

EU (1987b), Council Regulation (Euratom) 3954/87 of 22 December 1987, laying down maximum permitted levels of radioactive contamination of foodstuffs and of feeding stuffs following a nuclear accident or any other case of radiological emergency. *EU Official Journal* L371, 30 December 1987, p. 11, Brussels.

FAO-WHO (1991), *Codex Alimentarius*, *Vol. 1 Section 6.1, Levels for Radionuclides*. Codex Alimentarius Commission, Geneva.

IAEA (1986), *Convention on Early Notification of a Nuclear Accident* and *Convention on Assistance in the Case of a Nuclear Accident or Radiological Emergency.* Legal Series No. 14, IAEA, Vienna.

IAEA (1996a), *Defence in Depth in Nuclear Safety*. INSAG-10, IAEA, Vienna.

IAEA (1996b), *International Basic Safety Standards for Protection against Ionizing Radiation and for the Safety of Radiation Sources*, Safety Series No. 115, IAEA, Vienna. Co-sponsored by FAO, IAEA, ILO, OECD NEA, PAHO and WHO.

IAEA (2002), *Preparedness and Response for a Nuclear or Radiological Emergency*, Safety Standards GS-R-2, IAEA, Vienna.

IAEA (2003a), *Method for Developing Arrangements for Response to a Nuclear or Radiological Emergency*. EPR-METHOD, IAEA, Vienna.

IAEA (2003b), *Application of Simulation Techniques for Accident Management Training in Nuclear Power Plants*, TECDOC-1352, IAEA, Vienna.

IAEA (2006), *Fundamental Safety Principles,* Safety Standards, SF-1, IAEA, Vienna.

IAEA (2007), *Arrangements for the Preparedness for a Nuclear or Radiological Emergency*, GS-G-2.1, IAEA, Vienna.

IAEA (2009), *The International Nuclear Event Scale (INES) User's Manual,* 2008 edition. IAEA, Vienna.

ICRP (1991), *1990 Recommendations of the International Commission on Radiological Protection,* ICRP Publication 60, Pergamon Press, Oxford and New York.

ICRP (1993), *Principles for Intervention for Protection of the Public in a Radiological Emergency*, ICRP Publication 63, Pergamon Press, Oxford and New York.

Martincic, R. and Obrentz, L. (2008), ConvEx-3 2008, 43-hour Global Drill. Lessons were learnt from a large-scale nuclear emergency exercise held in July that tested international readiness. *IAEA Bulletin*, 50(1), Vienna.

NEA (2007), *Strategy for Developing and Conducting Nuclear Emergency Exercises*. Nuclear Energy Agency, No. 6162, OECD, Paris.

13
Non-proliferation safeguards in nuclear power programmes

M. S. PELLECHI, International Atomic Energy Agency (IAEA), Austria

Abstract: This chapter explores non-proliferation from the point of view of international safeguards and recommends what 'newcomers' should be familiar with if they are to successfully assess, manage or participate in the expanded use of nuclear energy. It provides a basic understanding of the safeguards requirements to be addressed by stakeholders, and offers some technical guidance and advice on safeguards-relevant operational measures that may be taken. The subject matter is presented in simplified terms, such that it may be of particular benefit to stakeholders with limited or no nuclear energy experience.

Key words: International Atomic Energy Agency, IAEA, Nuclear Non-Proliferation Treaty, NPT, safeguards, non-proliferation, safeguards agreement, additional protocol, state system of accounting for and control of nuclear material, SSAC.

13.1 Introduction

The Treaty on the Non-Proliferation of Nuclear Weapons (otherwise known as the Nuclear Non-Proliferation Treaty or NPT) was brought into force in part out of a desire to contain the spread of nuclear weapons and nuclear weapons technology, while legitimizing the peaceful uses of nuclear energy. The text of the NPT can be found in INFCIRC/140 (IAEA, 1970). From a global perspective, an increasing number of countries are today assessing, or plan to include, the use of nuclear power as part of the mix of sustainable energy sources. According to Amano (2010), the Director General of the International Atomic Energy Agency (IAEA), in excess of 20 countries

Published by Woodhead Publishing Limited, 2012

might very well bring their first nuclear power plant online within the next 20 years. Towards that end, the IAEA, one of the specialized agencies[1] of the United Nations (UN), has established a website dedicated to helping Member States develop a nuclear power infrastructure. Readers of this chapter may want to familiarize themselves with some of the authoritative publications, specifically IAEA (2007a) and IAEA (2008a), as the information contained in them will assist in gaining an understanding of where safeguards fits into the development of a State's nuclear power infrastructure. An overview of the IAEA Safeguards System can be found in footnote.[2]

If, as projected, any manner of a nuclear renaissance is realized, it is expected that some of these States will be developing countries. And therein arises a necessity for the safeguarding of nuclear material and facilities in countries that previously had very limited or no experience with the nuclear fuel cycle and international safeguards. As indicated in the IAEA (2007a) 'Milestones' publication, it is essential for all concerned stakeholders to understand the safeguards requirements and obligations, in addition to the other 18 topical areas requiring commitment and resources.

This chapter's objective is to provide guidance to stakeholders with an understanding of what is needed for the effective implementation of safeguards, when it is needed, and how, through the transparent application of safeguards, they may advance their interests in the peaceful use of nuclear energy nationally and internationally. It begins with a discussion on the underlying safeguards requirements as they derive from the NPT. The chapter examines, in general terms, the international non-proliferation obligations of countries/stakeholders within the context of a comprehensive safeguards agreement (IAEA, 1972) and additional protocol (IAEA, 1997a). Together with examples of the application of safeguards measures, the chapter explores the establishment of an effective state system of accounting for and control of nuclear material, and offers some technical perspective on the NPT and the IAEA. It also provides a brief discussion on transparency and the future of safeguards.

Each subsection of the chapter is self-contained which, while building on the previous subsection(s), can be read in a stand-alone fashion for quick reference. Nevertheless, an underlying theme throughout the subsections is that stated intentions alone are not enough to assure the global community

[1] A specialized agency refers to an autonomous organization linked to the UN through special agreements. A current listing of such agencies may be viewed at http://www.un.org/Overview/uninbrief/institutions.shtml

[2] The Safeguards System of the International Atomic Energy Agency, available from http://www.iaea.org/OurWork/SV/Safeguards/safeg_system.pdf

that any new pursuit or expansion of a civilian nuclear option is entirely for peaceful purposes. The chapter is written with the presumption that it is primarily through demonstrable, transparent actions by prospective governments and nuclear facility operators that a country convinces its stakeholders that their efforts represent a positive, peaceful use of nuclear material and technology.

A short glossary of frequently used terms is provided below (IAEA, 2001, and IAEA Statute, Article XX: Definitions):

- *Additional Protocol (AP):* A protocol additional to a safeguards agreement (or agreements) concluded between the IAEA and a State, or group of States, following the provisions of the Model Additional Protocol. The Model Additional Protocol provides for those measures for strengthening the effectiveness and improving the efficiency of IAEA safeguards which could not be implemented under the legal authority of safeguards agreements.
- *Comprehensive Safeguards Agreement (CSA):* An agreement that applies safeguards on all nuclear material in all nuclear activities in a State.
- *Facility:* A reactor, a critical facility, a conversion plant, a fabrication plant, a reprocessing plant, an isotope separation plant or a separate storage installation; or any location where nuclear material in amounts greater than one effective kilogram is customarily used.
- *Location Outside Facilities (LOF):* Any installation or location, which is not a facility, where nuclear material is customarily used in amounts of one effective kilogram or less.
- *Nuclear material:* Any source material or special fissionable material as defined in Article XX of IAEA Statute.
- *Source material:* Uranium containing the mixture of isotopes occurring in nature; uranium depleted in the isotope 235; thorium; any of the foregoing in the form of metal, alloy, chemical compound, or concentrate; any other material containing one or more of the foregoing in such concentration as the Board of Governors shall from time to time determine; and such other material as the Board of Governors shall from time to time determine.
- *Special fissionable material:* Plutonium-239; uranium-233; uranium enriched in the isotopes 235 or 233; any material containing one or more of the foregoing; and such other fissionable material as the Board of Governors shall from time to time determine; but the term 'special fissionable material' does not include source material.
- *Uranium enriched in the isotopes 235 or 233:* Uranium containing the isotopes 235 or 233 or both in an amount such that the abundance ratio

of the sum of these isotopes to the isotope 238 is greater than the ratio of the isotope 235 to the isotope 238 occurring in nature.

13.2 Nuclear Non-Proliferation Treaty (NPT)

13.2.1 Birth of a landmark treaty

On Monday, 16 July 1945, the world's first nuclear weapon was detonated by the United States. By the mid-1960s, there were five States which had produced and tested nuclear weapons, including China, France, the former Soviet Union (USSR; today called the Russian Federation), the United Kingdom (UK) and the United States (US). Recognizing the negative impact to their respective national interests if other States were to produce and test such devices, two of the nuclear-weapon States, the US and the USSR, sought to erect an institutional mechanism to limit the further spread of nuclear weapons or other nuclear explosive devices. The Treaty on the Non-Proliferation of Nuclear Weapons (NPT) was first adopted on 12 June 1968, and then on 5 March 1970 it was brought into force. There are two distinct but interrelated NPT verification goals: to build confidence between parties; and to deter against treaty violation by risk of detection.

The text of the NPT segregates the signatories into two camps: the 'haves' (i.e., nuclear-weapon States, NWS) and the 'have-nots' (i.e., non-nuclear-weapon States, NNWS). The Treaty defines a NWS as one which has manufactured and exploded a nuclear weapon or other nuclear explosive device prior to 1 January 1967. This meant that five States were recognized as declared nuclear-weapon States at the time the Treaty entered into force: China, France, the USSR, the United Kingdom and the United States. Since 1 January 1967, several other States are known to have, or are assumed to have, conducted a nuclear weapons test. These countries are not recognized as nuclear-weapon States according to the NPT's definition.

The NPT was given an initial 25-year lifespan in Article X of the Treaty, though another provision, Article VIII, entails a review process that occurs every five years with the goal of assuring that the Treaty's objectives are being realized. During the 1995 NPT Review and Extension Conference, a decision that the NPT shall continue in force indefinitely was included among the package of decisions that were adopted.[3] The Conference also reaffirmed the universality of the NPT, stating 'Universal adherence to the Treaty on the Non-Proliferation of Nuclear Weapons is an urgent priority. All States not yet Party to the Treaty are called upon to accede to the Treaty

[3] Official documents on the package of three decisions in all languages are available from the United Nations Office for Disarmament Affairs (UNODA) from http://www.un.org/disarmament/WMD/Nuclear/1995-NPT/1995NPT.shtml

at the earliest date, particularly those States that operate unsafeguarded nuclear facilities.' The most recent NPT Review Conference was held in 2010 and concluded with the adoption of a 22-point action plan (over the next five years) to advance the three main pillars of the Treaty: nuclear disarmament, non-proliferation and the peaceful uses of nuclear energy. The final 2010 document also provides an article-by-article review of the NPT's operations, taking into account the decisions and resolutions previously adopted by both the 1995 Review and Extension Conference and the 2000 Review Conference.[4]

13.2.2 NPT and regional treaties

Under Article VII of the NPT, it is specifically recognized that a group of States have a right to conclude regional treaties 'in order to assure the absence of nuclear weapons in their respective territories'. Thus, when a NNWS is giving consideration to building its first nuclear power plant, as part of the decision process they should also consider the impact that any relevant regional treaty will have.

Today, there are a number of regional treaties dealing with nuclear-weapon-free zones, each of which obligates the Parties to conclude a comprehensive safeguards agreement with the IAEA. Examples of regional treaties with the goal of establishing a nuclear-weapon-free zone include the Treaty for the Prohibition of Nuclear Weapons in Latin America (Tlatelolco Treaty)[5], the African Nuclear-Weapon-Free Zone Treaty (Pelindaba Treaty)[6], the Southeast Asia Nuclear-Weapon-Free Zone Treaty (Treaty of Bangkok)[7], the Central Asian Nuclear-Weapon-Free Zone

[4] The Final Document of the 2010 Review Conference consists of four parts in three volumes. The Final Document of the 2010 NPT Review Conference (Parts I and II) NPT/CONF.2010/50 (Vol. I) is available from http://www.un.org/ga/search/view_doc.asp?symbol=NPT/CONF.2010/50 (VOL.I); relevant documents and conference working papers, decisions, and notes are published at http://www.un.org/en/conf/npt/2010/docs.shtml

[5] The Tlatelolco Treaty, which entered into force on 25 April 1969 (before the NPT was in force), obligated that 'Each Contracting Party shall negotiate multilateral or bilateral agreements with the International Atomic Energy Agency for the application of its safeguards to its nuclear activities'.

[6] The Pelindaba Treaty, which entered into force on 15 July 2009, obligated that 'Each Party undertakes . . . to conclude a comprehensive safeguards agreement with IAEA . . .'.

[7] The Treaty of Bangkok, which entered into force on 27 March 1997, obligated that 'Each State Party which has not done so shall conclude an agreement with the IAEA for the application of full scope safeguards to its peaceful nuclear activities . . .'.

Treaty (CANWFZ or Treaty of Semipalatinsk)[8] and the South Pacific Nuclear-Free Zone Treaty (Rarotonga Treaty).[9]

13.2.3 Non-nuclear-weapon states as stewards of nuclear material and technologies

Of particular safeguards interest and importance to NNWS are the treaty provisions contained in Articles II, III, IV and VI of the NPT.[10] For simplification purposes, by its sovereign decision to accede to the NPT, all NNWS Parties to the Treaty commit not to directly or indirectly receive, manufacture or otherwise acquire any nuclear weapons or other nuclear explosive devices, including receiving assistance in the development and manufacturing of such devices.[11] Additionally, all NNWS Parties to the NPT are legally bound to accept safeguards on all source or special fissionable material in all peaceful nuclear activities within the territory of such State, under its jurisdiction, or carried out under its control anywhere in accordance with a safeguards agreement to be negotiated and concluded with the IAEA (in accordance with the Statute of the IAEA and the IAEA's safeguards system).[12] These comprehensive safeguards agreements (CSA) are modelled on the standard agreement INFCIRC/153 (Corrected) published in IAEA (1972).

In exchange for the above commitments, all States party to the NPT affirm that the principle benefits of peaceful application of nuclear technology is an 'inalienable right of all the Parties to the Treaty'[13], and that the States shall undertake negotiations on effective measures for nuclear arms reductions with the goal of eliminating all nuclear weapons (i.e., nuclear disarmament).[14]

[8] The CANWFZ Treaty, which entered into force on 21 March 2009, obligated that 'Each Party undertakes . . . to conclude with the IAEA and bring into force, if it has not already done so, an agreement for the application of safeguards in accordance with the NPT (INFCIRC/153 (Corr.)), and an additional protocol (INFCIRC/540 (Corr.))'.

[9] The Rarotonga Treaty, which entered into force on 11 December 1986, obligated that 'The agreement referred to in paragraph 1 shall be, or shall be equivalent in its scope and effect to, an agreement required in connection with the NPT on the basis of the material reproduced in document INFCIRC/153 (Corrected) of the IAEA. Each Party shall take all appropriate steps to ensure that such an agreement is in force . . .'.

[10] The aforementioned provisions and their relevance to the implementation of safeguards in a NNWS are covered in detail in Section 13.4, Non-proliferation responsibilities.

[11] Ref. Article II of the NPT.

[12] Ref. Article III of the NPT.

[13] Ref. Article IV of the NPT.

[14] Ref. Article VI of the NPT.

13.2.4 Nuclear-weapon states as stewards of nuclear material and technologies

In the case of the five NPT declared NWSs, the provisions contained in Articles I, III, IV and VI of the NPT are of direct relevance to the issue of safeguards. For example, each NWS undertakes '. . . not in any way to assist, encourage, or induce any non-nuclear-weapon State to manufacture or otherwise acquire nuclear weapons or other nuclear explosive devices, or control over such weapons or explosive devices'.[15] NWSs are also obliged not to provide any NNWS with source or special fissionable material, or equipment or material especially designed or prepared for the processing, use or production of special fissionable material unless the material is subject to the safeguards.[16]

In exchange for the commitments made by the NNWS Parties to the NPT, the NWSs affirm that the NWS shall undertake negotiations on effective measures for nuclear arms reductions with the goal of eliminating all nuclear weapons (i.e., nuclear disarmament).[17]

Though they are not required to have a safeguards agreement with the IAEA, each NWS has chosen to do so. A NWS's safeguards agreement with the IAEA is referred to as a Voluntary Offer Agreement (VOA).[18] The IAEA recognizes that VOAs serve two purposes: to 'broaden the IAEA's safeguards experience at advanced facilities, and to demonstrate that nuclear-weapon States are not commercially advantaged by being exempt from safeguards on their peaceful nuclear activities', as explained in IAEA (2007b), page 7. In practice, the safeguards measures implemented in accordance with VOAs are only applied with regard to declared nuclear material in selected facilities in one or more of the five States.

13.3 International Atomic Energy Agency (IAEA) and international safeguards

Today, the IAEA has safeguards agreements in force with over 170 countries around the world. Almost all of these agreements are formulated

[15] Ref. Article I of the NPT.
[16] Ref. Article III of the NPT.
[17] Ref. Article VI of the NPT.
[18] Each VOA generally follows the format of agreements based on INFCIRC/153 (Corr.) but varies in the scope of nuclear material and facilities covered. For example, such VOAs exclude material and facilities with national security significance, and foresee the possibility of withdrawing such material and facilities from safeguards.

based on INFCIRC/153 (Corrected) (IAEA, 1972) in respect of a State's obligation for a CSA as a Party to the NPT.[19]

13.3.1 'Atoms for peace'

In August 1945, shortly after the June 1945 signing of the UN Charter by the Heads of State, two atomic bombs were dropped on the Japanese cities of Hiroshima and Nagasaki, bringing an end to World War II. Subsequently, fears arose that atomic weapons could spread, and with them the potential for mass destructive power never before seen on such a scale.

With international attention focused on the atom, on 8 December 1953 before the 470th Plenary Meeting of the UN General Assembly, US President Eisenhower delivered an address titled 'Atoms for peace'.[20] During the course of his speech, he stated:

> I therefore make the following proposal. The governments principally involved, to the extent permitted by elementary prudence, should begin now and continue to make joint contributions from their stockpiles of normal uranium and fissionable material to an international atomic energy agency. We would expect that such an agency would be set up under the aegis of the United Nations.

13.3.2 Statute of the IAEA

In April 1955, work had began on drafting the Statute of the IAEA with the participation of governmental representatives from Australia, Belgium, Canada, France, Portugal, South Africa, the UK, and the US. Later, in early 1956, the group expanded to include representatives from Brazil, the former Czechoslovakia, India, and the USSR. These historical events have been summarized in IAEA (1997b).

As well described by Fischer (2003), the IAEA's founders held the view that there were three primary functions for the new Agency, namely:

[19] For completeness, a safeguards agreement based on INFCIRC/153 (Corrected) is not the only basis for safeguards application in a State by the IAEA. Safeguards have been implemented between the IAEA and States based on other types of agreements as well. For example, the three non-NPT States (India, Israel and Pakistan) have in force item-specific safeguards agreements based on the IAEA's safeguards system approved by the Board of Governors in 1965 and extended in 1966 and 1968 as set forth in INFCIRC/66/Rev.2 published in IAEA (1968). These item-specific agreements have provided for the application of safeguards to nuclear material, specified items (e.g. heavy water, nuclear-related equipment, zirconium tubes) and facilities. The five NPT declared nuclear weapon States have a Voluntary Offer Agreement or VOA, in force as noted in Section 13.2.4, Nuclear-weapon states as stewards of nuclear material and technologies.

[20] The full text of US President Eisenhower's speech to the General Assembly is available from http://www.iaea.org/About/history_speech.html

1. To promote the peaceful use of nuclear energy throughout the world
2. To ensure that any nuclear plant, activity or information it works with, is used only for peaceful purposes
3. To ensure the safe use of any such plant, activity or information.

This perspective took root during the development of the IAEA Statute, which was formally approved on 23 October 1956 by the Conference on the Statute of the IAEA, held at the Headquarters of the UN.[21] Eighty-one nations voted unanimously to approve the IAEA Statute. Thereafter, the IAEA Statute entered into force 29 July 1957, by which time 26 States had deposited their instruments of ratification. Thus, the IAEA was established as an autonomous organization, independent of the UN through its own international treaty, the IAEA Statute; however, the IAEA reports to both the UN General Assembly and the UN Security Council.[22] In this regard, the IAEA's relationship with the UN is regulated by special agreement dated 30 October 1959 (reproduced in INFCIRC/11) (IAEA, 1959). Organizationally, the IAEA comprises a Secretariat, headed by a Director General, together with two policy-making bodies: the 35-member Board of Governors and the General Conference which consists of all Member States.

On 29 July 2007, the IAEA officially turned 50. During the interim years, its Statute has been amended several times.[23]

13.3.3 Safeguards and verification

Since its birth, the IAEA's safeguards work has included[24]:

- Verifying that countries are not using nuclear material and nuclear technology for non-peaceful purposes
- Setting the standards and guidelines for safeguarding nuclear material and facilities
- Fulfilling its role as the guardian of the nuclear non-proliferation treaty
- Assisting the international community in nuclear disarmament efforts.

To fulfil its mandate, the Secretariat has assigned the Department of Safeguards with organizational responsibility for safeguards implementation. In practical terms, the Department of Safeguards implements monitoring and verification activities worldwide in over 900 facilities and locations

[21] The complete text of the IAEA Statute is available from http://www.iaea.org/About/statute_text.html
[22] Ref. Note 3 on web page at http://www.un.org/en/aboutun/structure/
[23] For amendments to the IAEA Statute, see the IAEA webpage from http://www.iaea.org/About/statute_amendments.html
[24] A historically definitive account of the IAEA is provided in Fischer (1997).

outside facilities (LOFs).[25] While covering the entirety of a State's nuclear fuel cycle, its efforts consider the strategic value of those types of nuclear material and activities in a NNWS that are the most crucial and relevant to nuclear weapons manufacturing. It publishes the results of its activities annually in a Safeguards Implementation Report.[26]

Keeping in mind that there are three major infrastructure milestones[27] in the Milestones publication (IAEA, 2007a) for the development of a nuclear power programme, stakeholders involved with any of the milestones may find themselves better equipped and more effective in their assigned capacity when they consider the full scope and nature of the IAEA's safeguards work. This subsection briefly discusses that work in the context of the Safeguards and Verification Pillar.[28]

Safeguards have evolved from their early focus on safeguarded nuclear material at the facility level, to today's concept which reflects development and implementation of a safeguards approach at the State level.[29] In the evolution of the safeguards approach over the last several decades, including transitioning the application of safeguards from a facility level to the State level, there were external and internal IAEA factors driving the introduction of new tools and safeguards measures. For an authoritative account, two recommended sources of information on the IAEA's historical transition in safeguards and how it was implemented and is being implemented today have been provided by Jennekens (1970) and by an IAEA Department of Safeguards document covering 1991–2005 (IAEA, 2005b, Section C).

For introductory purposes, the general sequence of developments in a State's safeguards system will be described starting from the moment a

[25] A short list of common safeguards terms such as facility and LOF is provided on pages 423–424 while a more extensive glossary can be found in IAEA (2001).
[26] The Safeguards Implementation Report published annually by the IAEA has a restricted distribution; however, a summary of the report is made available to the public. For the latest publicly available Statement, see the link titled 'Latest Statement and Background' on the IAEA's web at http://www.iaea.org/OurWork/SV/Safeguards/index.html
[27] Defined as Milestone 1: Ready to make a knowledgeable commitment to a nuclear programme; Milestone 2: Ready to invite bids for the first nuclear power plant; Milestone 3: Ready to commission and operate the first nuclear power plant.
[28] Three main pillars underscore the mission of the IAEA: Safeguards and Verification; Safety and Security; Science and Technology.
[29] The IAEA utilizes a State-level concept for the implementation and evaluation of safeguards. The State-level concept is the IAEA's holistic approach to safeguards implementation which is applicable to all States. It is based on a comprehensive State evaluation process and a State Level Approach (SLA), implemented through an Annual Implementation Plan (AIP) for a State.

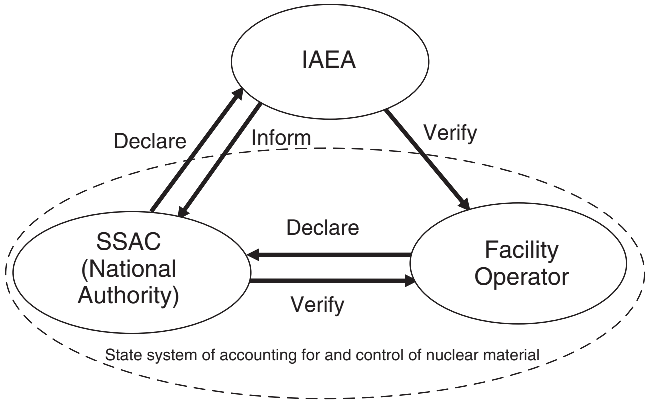
State decides to pursue a nuclear energy option for peaceful purposes. In terms of a chronology of events, what follows is a hypothetical sequence of safeguards-relevant events that a 'newcomer' State may come to experience. Bear in mind that the sequence described below is for illustrative purposes only, and there may be many variations in an actual case.

In theoretical terms, assuming a State has not yet acceded to the NPT, it may take appropriate action domestically and internationally for considering such an undertaking. Should the State be contemplating development of a nuclear power infrastructure, the State will very likely assess its priorities and presumably decide positively on the importance of the NPT relative to its non-proliferation goals and objectives. Whether a State accedes to the NPT or not, their decision will inevitably impact the perception of many regional and international stakeholders regarding the State's transparency and openness.

For a State without a safeguards agreement in force, the IAEA will work closely with the State's representatives to draft and bring one into force. If the reference State is a Party to the NPT, the safeguards agreement will be comprehensive in nature and modelled on INFCIRC/153 (Corrected) (IAEA, 1972). Any State with a safeguards agreement may also conclude a protocol additional to its safeguards agreement with the IAEA (hereinafter referred to as an 'additional protocol').[30] While it remains the sovereign decision of the host State, if requested the IAEA will advise and provide available support to enable a State to conclude an additional protocol (AP) to their safeguards agreement at the same time (or at a later date depending on the State's circumstance). In respect of drafting the safeguards agreement (and AP), the State and the IAEA will conclude Subsidiary Arrangements[31] which specify how the measures included in the CSA (and AP) are to be implemented. Subsidiary Arrangements to

[30] Following the IAEA Board of Governors approval in 1997, the IAEA began concluding with States having a safeguards agreement an additional protocol based on the provisions of a standard reproduced in INFCIRC/540 (Corrected) (IAEA, 1997a). It is a binding instrument complementary to the CSA that a State may voluntarily undertake. By doing so, a State undertakes additional commitments in addition to those in the CSA. The protocol is discussed in Section 13.4.2, Model additional protocol. Additional protocols to item-specific safeguards agreements and voluntary offer agreements may also be concluded with the IAEA.

[31] Subsidiary Arrangements are defined by the IAEA as 'the document containing the technical and administrative procedures for specifying how the provisions laid down in a safeguards agreement are to be applied. Under an INFCIRC/153-type safeguards agreement, the State and the IAEA are required to agree on Subsidiary Arrangements. Under an additional protocol to a safeguards agreement (or agreements), if either the State or the IAEA indicates that Subsidiary Arrangements are necessary, then both parties are required to agree on such Arrangements.'



13.1 IAEA–State–facility relationship (adapted from IAEA training material).

safeguards agreements consist of a General Part, applicable to all common nuclear activities of the State concerned, and a Facility Attachment, prepared for each facility in the State and describing the arrangements specific to that facility.

Once a CSA is in force, the State is obligated to establish a state system of accounting for and control of nuclear material (SSAC) if it has not already done so.[32] In this regard, there are two different contexts to the use of the term SSAC. One refers to the National Authority assigned responsibility as the formal technical interface for safeguards implementation with the Agency and facility operators. The second refers to the system of nuclear material accounting and control procedures required by the National Authority and implemented by facility operators. The concept is graphically presented in Fig. 13.1. With regard to the National Authority assigned as the SSAC, among its many initial responsibilities, the newly established SSAC will address the preparation and transmission of the State's initial

[32] INFCIRC/153 (Corrected) (IAEA, 1972) states, '. . . the State shall establish and maintain a system of accounting for and control of all nuclear material subject to safeguards under the Agreement, and that such safeguards shall be applied in such a manner as to enable the Agency to verify, in ascertaining that there has been no diversion of nuclear material from peaceful uses to nuclear weapons or other nuclear explosive devices, findings of the State's system. The Agency's verification shall include, *inter alia*, independent measurements and observations conducted by the Agency in accordance with the procedures specified in Part II below. The Agency, in its verification, shall take due account of the technical effectiveness of the State's system.'

report (concerning the inventory of nuclear material and facilities) to the IAEA.[33] If an AP is in force at the time, the preparation and submission of initial AP declarations are also required.

Early consultation with the IAEA can facilitate this process to the benefit of both Parties. Experience shows that it is a good practice if the SSAC is involved early in the process, for example prior to or during the development of the Subsidiary Arrangements. In some NNWSs, it is an office in the Ministry of Foreign Affairs or equivalent that is designated as the SSAC, negotiates the Subsidiary Arrangements, submits the initial report and undertakes to fulfil other SSAC responsibilities. In other countries, a department or section in the Ministry of Science and Technology or equivalent is the SSAC which seeks to fulfil the requisite obligations. There are any number of considerations that go into such a decision and each State is expected to factor in its own national interests and needs. Examples may include current or proposed national laws, policies and regulations relevant to safeguards implementation; relevant foreign policies; technical competency of the ministry or department/section; and financial and other resource availability/constraints. The essential point here is that whether a State has no nuclear material and facilities, or they possess a more developed nuclear fuel cycle, having a technically capable and properly resourced SSAC is a requirement of fundamental importance. Regardless of which national authority is designated as the SSAC, the State's point of contact for safeguards will be identified in the relevant Subsidiary Arrangements and will benefit by being involved early in the process.

With regard to the system of nuclear material accounting and control procedures, if there are small quantities of existing nuclear material in the State, the State may already have an operational nuclear material accounting and control system. If one is not in place, concurrent with the activities involving the preparation of the initial inventory declaration, the State and the relevant operators establish or implement a nuclear material accounting and control system.[34] In keeping with the IAEA's mandate under its Statute, the IAEA takes account of all safeguarded nuclear material, including enriched uranium, plutonium and uranium-233 in countries with a CSA. Other types of nuclear material subject to safeguards verification include thorium, natural uranium, and depleted uranium, the latter of which is

[33] In a CSA based on INFCIRC/153 (Corrected), the State has an obligation to declare of all its nuclear material and facilities in peaceful activities to the IAEA.
[34] This will include, for example, the following areas as relevant: accounting and control of starting point of safeguards; transfers, terminations and exemptions/de-exemptions; accidental losses and gains; categorization of nuclear material; material balance areas; and records and reports system.

commonly used, for instance, as shielding of radiation sources in hospitals, industry and agriculture.[35]

Following the submission, if any nuclear material is declared, the SSAC works closely with relevant nuclear operators to prepare for the IAEA's initial verification of the State's nuclear material inventory. During this period, the IAEA is reviewing the State's declarations and, as appropriate, assessing the correctness and completeness of the submittal(s). In some instances, the SSAC may receive one or more IAEA requests for clarification or further information. The more technically capable a SSAC is at this point, the more easily it will be able to respond accordingly. Good communication between the parties is always a recommended priority, but it is especially important during this stage of the process.

In respect of the State's initial inventory declaration, the SSAC will then undertake to routinely submit accounting and operating reports to the IAEA as specified in the respective Facility Attachment (part of the Subsidiary Arrangements). The relevant accounting reports and operating records in the submittals usually originate from the facility operator's system of accounting for and control of nuclear material. Nevertheless, as the CSA is a binding instrument between the IAEA and the State, it is the SSAC that is responsible for assuring the correctness and completeness of the submittals to the IAEA. This point is very important, and essentially serves as a reminder to those responsible in understanding and fully embracing their SSAC role as the technical interface to the IAEA.

In the case of an existing or planned nuclear facility, the SSAC will typically consult with the operator(s) for the preparation of a design information package that is to be submitted to the IAEA. The design information, in the form of a Design Information Questionnaire (DIQ) for each existing and planned nuclear facility, is used by the IAEA, together with the information provided in the State's declaration of the initial inventory of nuclear material, to facilitate the development of the safeguards approach at the State level, including the safeguards measures to be implemented at the respective nuclear facility. Where applicable, the facility design information is also used in (1) development of the Facility Attachment, (2) technical discussions between the IAEA, the SSAC and the facility operator regarding the potential installation and service of IAEA containment and surveillance (C/S) systems, and (3) the IAEA's ongoing assessment concerning the identification of indicators of misuse of declared facilities and/or diversion of declared nuclear material from peaceful activities.

The application of C/S systems is complementary to the accounting measures implemented in accordance with the State's safeguards agreement. As

[35] Undeclared irradiation of fertile material (thorium-232 or uranium-238) could be carried out to produce fissile material (U-233 or Pu-239).

C/S devices make use of the local design features of the facility, equipment or item (e.g., nuclear material in a storage container or vault), their application is dependent on a number of factors. Such factors include defining the objective of applying such safeguards measures (e.g., whether it is an indicator of possible diversion of nuclear material, an indicator of possible misuse of sensitive equipment or process, or an indicator of possible tampering with IAEA safeguards equipment). Other factors include, but are not limited to, the type and form of nuclear material (to the extent applicable), design information on the containment equipment or facility, and alternative safeguards measures to meet the same objective. As a practical matter, C/S systems are typically utilized in situations where they offer improved safeguards efficiencies or effectiveness, contribute to improvements in personnel safety, health and radiation protection, or are attached to IAEA equipment and other sensitive items (e.g., a sealable pouch containing facility design information) to provide an indication of possible tampering.

For example, we hypothetically consider a large number of containers of nuclear material containing low-enriched uranium oxide powder in a separate storage area of a facility, where they are to be stored for many years before final disposition. The first time a physical inventory verification is conducted at the facility, the time-consuming task of performing detailed measurements and assays of the nuclear material in the containers will be conducted. During subsequent physical inventory verifications, some of the inspection activities might be reduced or eliminated by the application of an appropriate IAEA C/S measure(s). In the example provided, installation and use of the containment (or surveillance) system is based in part on the cost-effectiveness of the approach. When considering the application of such measures, the IAEA will consult the State in advance of any installation, and they will jointly decide on the merits of any increased efficiencies to be achieved.

If an AP is also in force, normally the State could expect that the first time a complementary access is requested is after the State's initial AP declarations are received and reviewed by the IAEA. In practice, after the initial AP declarations are submitted, AP update declarations are routinely sent by the State and the majority of complementary accesses conducted are to assure the absence of undeclared nuclear material and activities at the selected location(s). From time to time, there may be 'questions' or 'inconsistencies' that arise. Experience shows that when the responsible national authority (e.g., SSAC if so designated) consults early and works closely with the IAEA, especially to resolve the questions or inconsistencies in a timely manner, then assurances that a State's nuclear programme is entirely for peaceful purposes can be strengthened. In this regard, the State's non-proliferation objectives are being achieved.

In time, the SSAC personnel could expect to include the following functional activities in their routine (i.e., day-to-day) safeguards-related activities:

- Periodic conduct of physical inventory takings of safeguarded nuclear material
- Provision of accounting reports and operating records to IAEA in accordance with the Safeguards Agreement and the relevant Subsidiary Arrangements
- Provision of IAEA inspector and technician access to relevant locations and strategic points
- Provision of support (e.g., availability of crane operator, refuelling bridge) during the conduct of on-site safeguards activities (e.g. inspections, C/S installation and maintenance)
- Provision of information and close consultation with the IAEA as appropriate to resolving inconsistencies or open issues
- Conduct of follow-up actions associated with resolution of discrepancies and open anomalies
- Training to maintain or enhance technical skills and abilities of the SSAC and operator personnel
- Development and/or revision of organizational practices, standards, policies and procedures relevant to safeguards implementation
- Consultation in the drafting and updating of Subsidiary Arrangements
- Purchase and maintenance of SSAC-owned safeguards equipment.

With time and experience, the SSAC role becomes more familiar to the personnel involved, especially during the conduct of on-site safeguards activities (e.g., inspections, complementary accesses, design information verification visits).

13.3.4 Safeguards conclusions

The IAEA's goal has been, and remains today, to draw soundly based safeguards conclusions through effective and impartial implementation of safeguards agreements. In fact, the IAEA's safeguards conclusions regarding correctness and completeness of a State's declaration for States with comprehensive safeguards agreements in force depends on the extent to which the Agency is equipped to detect undeclared nuclear material and activities in such States. Under a safeguards system that is based on INFCIRC/153 (Corrected) (IAEA, 1972) alone, the IAEA is limited in its ability to assess undeclared nuclear material and activities. It is recognized that with the AP-related access provisions, availability of expanded State-declared information and broader access to locations in the State, the Agency's capability to detect and deter undeclared nuclear material or activities is significantly advanced.

When both a CSA and an AP are in force for a NNWS, and the IAEA finds that there is no indication of the diversion of declared nuclear material from peaceful activities, and no indication of undeclared nuclear material and activities for that State, the IAEA is able to draw a safeguards conclusion for the State that 'all nuclear material remained in peaceful activities'. However, if the evaluations regarding the absence of undeclared nuclear material and activities for a State remain ongoing as part of the State evaluation process, then the IAEA concludes for the State that 'declared nuclear material remained in peaceful activities'.

In those NNWSs where a CSA is in force alone (i.e., AP is not in force), based on the IAEA's findings that there is no indication of the diversion of declared nuclear material from peaceful activities in the State, the IAEA is able to draw a conclusion that the 'declared nuclear material remained in peaceful activities' for that State.

In the case of NNWS Parties to the NPT who have not yet brought comprehensive safeguards agreements with the IAEA into force as required by Article III of the NPT, the IAEA cannot draw any safeguards conclusions.

13.4 Non-proliferation responsibilities

The issue of proliferation extends beyond the NPT and the corresponding NNWS's obligation to accept safeguards on all source or special fissionable material and to undertake a comprehensive safeguards agreement with the IAEA. As this book is focused on infrastructure and methodologies for the justification of nuclear power programmes, the three milestones[36] described in the Milestones publication (IAEA, 2007a) are the focus of the detailed discussion on non-proliferation in this subsection. The information is organized by the essential obligation/commitment undertaken by a NNWS Party to the NPT, with associated safeguards requirements linked to the relevant milestone(s) in IAEA (2007a).

In generic terms, prior to reaching Milestone 1, the State is normally working to acquire a comprehensive understanding of the requisite obligations and commitments involved. Once a decision to proceed with the infrastructure development is made, the State organizes the national means and plans needed to successfully implement the decision while progressing towards Milestones 2 and 3. As a State advances with its nuclear energy plans, it would be beneficial for the State to periodically perform a self-assessment, keeping in mind some example metrics presented in Table 13.1.

[36] Milestone 1 is defined as when the State is ready to make a knowledgeable commitment to a nuclear power programme as it pertains to each of 19 issues, one of which is safeguards; Milestone 2 is defined as when a State is ready to invite bids for the first nuclear power plant; Milestone 3 is defined as when a State is ready to commission and operate its first nuclear power plant.

Table 13.1 Example State-level metrics for IAEA milestones: safeguards

Milestone no.			State-level metrics relevant to safeguards (to be achieved prior to reaching the identified milestone)
1*	2**	3***	
X			Understood the level of safeguards commitment required for the full life cycle of a nuclear power plant.
X			Established a plan or road map for safeguards implementation.
X			Committed to developing its nuclear power infrastructure transparently.
	X		Acceded to, or completed a decision process/plan for joining, appropriate international and regional legal treaties and conventions (e.g., Treaty on the Non-Proliferation of Nuclear Weapons).
	X		Concluded a comprehensive safeguards agreement with the IAEA conforming to INFCIRC/153 (Corrected), and where applicable, placing in force an additional protocol modelled on INFCIRC/540 (Corrected).
	X		Established a National Authority as the technical interface to the IAEA (i.e., SSAC) with the necessary authority, resources, and technical capability.
	X		Submitted, or in the process of submitting, requisite information to the IAEA in accordance with relevant safeguards obligations (e.g., initial report on inventory of nuclear materials and facilities, early provision of facility design information, AP-relevant declarations as applicable).
	X		Developed, or progressing in a programme plan for developing, a comprehensive framework covering all aspects of non-proliferation (e.g., nuclear-related import/export controls, use/ownership of nuclear material) including safeguards.
		X	Established a state system of accounting for and control of nuclear material which meets IAEA requirements.
		X	Concluded, or progressing in the development of, Subsidiary Arrangements with the IAEA, including relevant Facility Attachment(s).
		X	Established the organizational elements at the State and facility level with the responsibility to ensure the non-proliferation of nuclear materials and technologies in accordance with relevant legal instruments.
		X	Established, or progressing in the implementation of, an outreach programne for maintaining transparency of the nuclear power programme.

* Milestone 1 is defined as when the State is ready to make a knowledgeable commitment to a nuclear power programme as it pertains to each of 19 issues outlined in the IAEA (2007a) publication.
** Milestone 2 is defined as when a State is ready to invite bids for the first nuclear power plant.
*** Milestone 3 is defined as when a State is ready to commission and operate its first nuclear power plant.

13.4.1 Model comprehensive safeguards agreement

Milestone 1

The system of safeguards modelled on INFCIRC/153 (Corrected) (IAEA, 1972) is designed to provide assurance about the exclusively peaceful use of nuclear material within the territory of a State, under its jurisdiction or carried out under its control anywhere. Such a comprehensive safeguards agreement (CSA) applies safeguards on all source and special fissionable material in all peaceful activities. As expressed in the Appendix to IAEA (2005a), concluding a safeguards agreement (and/or additional protocol) with the IAEA generally requires two or three steps:

1. The State notifies the Agency of its intention to conclude a safeguards agreement and/or an additional protocol, and asks the Agency to submit the draft text(s) to the IAEA Board of Governors for approval, in order for the Board to authorize the Director General to sign and implement it. The notification should contain information on the applicable entry into force procedure (see step 3 below). The text(s) will then be submitted to the Board of Governors, which needs to authorize the Director General to sign, and will subsequently implement the agreement or protocol. After this, the documents are open for signature. Model letters are available from the IAEA (IAEA, 2008b, Annexes 1 and 2).
2. A representative of the State and the Director General sign the text(s). This may be done by the Head of State, Head of Government or Minister for Foreign Affairs or by any other government official – such as the Resident Representative to the Agency – with full powers to sign.
3. The State has two options to bring into force its safeguards agreement/protocol: either upon signature or on the date the Agency receives from the State written confirmation that its domestic requirements for entry into force have been met. If the latter option is applied, the third step required is for the State to provide such notification to the Agency. Again, a model letter is available from the IAEA (IAEA, 2008b, Annex 3).

From a State's point of view, there are two fundamental points that a 'newcomer' may want to keep in mind, especially when the NNWS is in the early stage of its considerations for launching a nuclear power programme (e.g., prior to Milestone 1):

1. By taking transparent actions, a State's national authority and nuclear facility operators convince their stakeholders that their efforts represent a positive, peaceful use of nuclear material and technology.

2. For the exclusive purpose of verifying fulfilment of a State's obligations assumed under the NPT,[37] the IAEA is the international authority vested with the right and obligation to ensure that safeguards are applied on all source and special fissionable material in all peaceful activities.[38]

In connection with a CSA, the IAEA applies safeguards with a general working hypothesis: non-compliance cannot be excluded and there is low but non-zero probability that a diversion can take place. In this respect, the objective of IAEA safeguards is the timely detection of diversion (of nuclear material) and deterrence through a risk of early detection. The IAEA achieves this objective by the tasks it performs and the safeguards measures that are implemented. For all NNWSs party to the NPT, the task is verifying the correctness and completeness of a State's declaration. Verifying the correctness of a State's declaration refers to providing meaningful assurance on the non-diversion of declared nuclear material, while verifying completeness of a State's declarations refers to providing credible assurance on the absence of undeclared nuclear material and activities.[39] From the analysis of all information available to it, including the results of the IAEA's field and headquarters activities, the IAEA derives safeguards conclusions that are reported annually to the Board of Governors in the Safeguards Implementation Report (SIR) for the previous calendar year.[40] Any cases of non-compliance with safeguards agreements are also reported in the SIR. The kinds of conclusion(s) that can be drawn depend upon the agreements that are in force.[41]

[37] Ref. Article III of the NPT.
[38] The application of comprehensive safeguards may also arise from other instruments, such as relevant regional treaties, bilateral agreements, or conditions of supply of nuclear-related items and technologies.
[39] The IAEA Board of Governors has affirmed that the scope of comprehensive safeguards agreements is not limited to nuclear material actually declared by a State, but includes any nuclear material that is required to be declared. That is, the Board confirmed that the IAEA has the right and obligation, under CSA-type agreements, to verify both the correctness (i.e., that the declaration includes the type(s) and quantity(ies) of the State's declared nuclear material holdings) and completeness (i.e., that everything is included that should have been declared) of the State's declaration.
[40] The IAEA's findings for a given State are recorded periodically in an internal document, known as a State Evaluation Report (SER). The report also includes any recommendations for follow-up action. The SER for such a State therefore includes the findings related to the correctness of the State's declaration. It also includes – to the extent possible in the absence of an additional protocol – the findings with regard to the completeness of those declarations.
[41] See Section 13.3.4 which explains the types of safeguards conclusions that may be drawn.

In this respect, it is essential for 'newcomer' States to readily understand the obligations arising from implementation of a Model Comprehensive Safeguards Agreement (INFCIRC/153 (Corrected)) (IAEA, 1972). In terms of a State's commitment, the reporting of nuclear material (e.g., all source and special fissionable material in all peaceful activities) and facilities includes:

1. Nuclear material which has reached the stage of processing where its composition and purity make it suitable for fuel fabrication or for isotopic enrichment
2. Export and import of material containing uranium or thorium which has not yet reached that stage of processing
3. Any nuclear material produced at a later stage
4. Any existing or planned nuclear facility.

As a matter of procedure, once a safeguards agreement based on INFCIRC/153 (Corrected) (IAEA, 1972) enters into force, the NNWS has an obligation to declare all of its nuclear material and facilities to the IAEA (referred to as a State's initial report).[42] The initial report (i.e., State declaration) is then verified by the IAEA and maintained on the basis of accounting reports submitted by the State and verification by the IAEA (for correctness and completeness).

Other important CSA-related obligations concern the State's commitment to:

- Establishing an effective system of accounting for and control of nuclear material
- Provision of timely access to the nuclear material, facilities and locations outside facilities[43]
- Provision of early design information for each nuclear facility (planned and existing).

[42] A State has the obligation to update this information and to declare all new nuclear material and facilities which subsequently becomes subject to the CSA. The status of each State's relevant safeguards agreement with the IAEA is available from http://www.iaea.org/OurWork/SV/Safeguards/sir_table.pdf. Other recommended Internet addresses: http://www.iaea.org/NewsCenter/News/2005/strengthening_sg.html and http://www.iaea.org/About/Policy/GC/GC49/Documents/gc49-9.pdf

[43] Facility is defined in INFCIRC/153 (Corrected) and INFCIRC/540 (Corrected) as: '(i) A reactor, a critical facility, a conversion plant, a fabrication plant, a reprocessing plant, an isotope separation plant or a separate storage installation; or (ii) Any location where nuclear material in amounts greater than one effective kilogram is customarily used'. Locations outside facilities are defined in INFCIRC/540 (Corrected) as 'any location, which is not a facility, where nuclear material is customarily used in amounts of one effective kilogram or less'.

Milestones 2 and 3

In verifying the correctness of a State's declaration, the IAEA applies nuclear material accountancy, complemented by containment and surveillance measures. As indicated earlier, in a safeguards regime based only on INFCIRC/153 (Corrected), the IAEA does not have all the tools necessary to fully assess the completeness of the State's declaration.[44] Despite the limitations, the IAEA does evaluate whether there are any indications of undeclared nuclear material and activities as part of a State evaluation process.[45] However, without the additional measures available under the AP, the IAEA remains unable to draw a conclusion on the absence of undeclared nuclear material and activities in the State as a whole.

Consequently, the focus of IAEA safeguards in these NNWSs has been to independently verify the correctness of the State's nuclear material accounting and operating records and reports that are maintained by the facility operators and the SSAC. This requires that the State provide accurate and complete declarations on all nuclear material and facilities/LOFs in order that appropriate safeguards measures can be implemented and relevant IAEA verification activities completed.

The IAEA verification activities for a NNWS with only a CSA in force are performed in accordance with prescribed requirements.[46] The technical requirements specify the activities considered necessary by the IAEA to provide a reasonable probability of detecting the diversion of a significant quantity of nuclear material from declared facilities and locations outside facilities. These IAEA verification activities are carried out during inspections and design information examination/verification visits.

Under a CSA, there are three types of inspections, each with defined IAEA access. In simple terms, these include:

1. *Ad hoc inspections*, which typically are made to verify a State's initial report on the nuclear material subject to safeguards, or reports on changes thereto, and to verify the nuclear material involved in international transfers. The IAEA's right of access is to any location where the

[44] Though safeguards strengthening measures implemented since the mid-1990s have increased the relative ability of the IAEA to detect undeclared nuclear material and activities, the activities that the Agency may conduct in this regard are limited for a State without an additional protocol.
[45] The State evaluation process is the IAEA's ongoing (i.e., continuous) approach to evaluation of all the information available to the Agency in exercising its rights and fulfilling its safeguards obligations.
[46] Technical safeguards criteria are established for each type of facility under safeguards and specify the scope, the normal frequency and extent of the verification activities needed to achieve the inspection goals at such facilities. They are used both for planning the implementation of verification activities and for evaluating the results arising from such activities.

initial report, or any inspections carried out in connection with it, indicate that nuclear material is present.

2. *Routine inspections*, which are carried out to verify the declared nuclear material and to verify the consistency of the Operator's records with the State's reports. These inspections may be conducted according to a defined schedule or they may be of an unannounced (or short-notice) character.[47] IAEA access is limited to strategic points defined in the relevant Subsidiary Arrangements and to locations with relevant records.
3. *Special inspections*, which may be carried out in circumstances according to defined procedures, if the IAEA considers that information made available by the State concerned, including explanations from the State and information obtained from routine inspections, is not adequate for the Agency to fulfil its responsibilities under the safeguards agreement. Access to information and/or to locations other than those specified for *ad hoc* and routine inspections may be made.

In addition to the above inspections, design information verification visits (called DIVs) to nuclear facilities may be conducted. Such visits would occur at appropriate times during the lifecycle of a declared facility, in order that the IAEA can verify the safeguards-relevant design information. For example, such visits may be carried out as follows:

- During construction of a nuclear power plant to determine the completeness of the declared design information
- Periodically during routine facility operations and/or following a plant maintenance outage to confirm that no safeguards-relevant modification was made to the NPP that would allow unreported activities to take place
- As part of a facility decommissioning to confirm that essential equipment was removed or rendered unusable.

The information above is intended to be a helpful addition to understanding the State's safeguards obligations referred to in Milestones 1–3 in the IAEA Milestones document (IAEA, 2007a).

13.4.2 Model additional protocol

Milestone 1

In response to developments in the 1990s (e.g., Iraq's clandestine nuclear weapons programme, South Africa's nuclear weapons programme), the IAEA developed and implemented additional, strengthened safeguards

[47] Paragraph 84 of INFCIRC/153 (Corrected) provides the IAEA with the right to conduct a portion of the routine inspections without advance notice. Under such a safeguards agreement, unannounced inspections are carried out in accordance with the principle of random sampling.

measures. Part of the strengthening measures approved by the IAEA Board of Governors and complementary to the safeguards agreement with a State is the 'Model Protocol Additional to the Agreement(s) between State(s) and the International Atomic Energy Agency for the Application of Safeguards' (INFCIRC/540 (Corrected)) (IAEA, 1997a). Following the IAEA Board of Governors approval in May 1997, the IAEA began concluding with those States that already had a safeguards agreement an additional protocol based on the provisions of a standard reproduced in INFCIRC/540 (Corrected).

When an AP is in force in a NNWS, the relevant articles of the AP oblige the State to provide additional information to the IAEA beyond that required under a CSA. The State's submissions are referred to as AP declarations. In very general terms, the AP declarations articulate all nuclear fuel cycle-related activities and facilities in the State (e.g., nuclear research and development not involving nuclear material, mining and milling), thereby going beyond the information required by the CSA (which is essentially focused on nuclear facilities and nuclear material). Beside the availability of broader information concerning a State's nuclear fuel cycle, the implementation of an AP in a State also permits the IAEA to access locations in the State or under its control (i.e., beyond declared nuclear facilities and LOFs) for any of the following purposes:

1. On a selective basis in order to assure the absence of undeclared nuclear material and activities
2. To resolve a question relating to the correctness and completeness of the information provided pursuant to the AP or to resolve an inconsistency relating to that information
3. For the Agency to confirm, for safeguards purposes, the decommissioned status of a facility or of a LOF where nuclear material was customarily used.

Due recognition should be given to the fact that it is a State's sovereign decision whether to place an AP in force, and as such, there are many important domestic and international considerations a State may have to address before it is willing to accept the relevant AP obligations. To date, over 100 States have an AP in force. These States have elected to do so in order to increase the effectiveness and efficiency of safeguards applied in the State and/or such action represents a continuing contribution to their international and national non-proliferation goals.

Milestones 2 and 3

It would be expected that once a State makes a decision to proceed with the infrastructure development (i.e., Milestone 1 is achieved), the State organizes the national means and plans needed to successfully implement its decision while progressing towards Milestones 2 and 3. If the State is contemplating whether to conclude an additional protocol to its safeguards agreement with the IAEA, it will necessarily want to understand the contents of the Model Additional Protocol (INFCIRC/540 (Corrected)) (IAEA, 1997a). Due to their direct relevance to drawing a safeguards conclusion, the relevant articles of the AP which cover the provision of information and complementary access will be discussed further.

Provision of information

Articles 2 and 3 of the AP specify the State's obligation for submitting timely declarations containing additional nuclear fuel cycle-related information. In very generic terms, Article 2 requires that the State provide the IAEA with declarations containing information on all parts of a State's nuclear fuel cycle – including that which is not required by the State's comprehensive safeguards agreement such as uranium mines, nuclear-related manufacturing locations, and nuclear waste sites – as well as any location where nuclear material is or may be present. This would include, for example, information on:

- Research and development activities related to its nuclear fuel cycle[48] including those not involving nuclear material
- All buildings (permanent and temporary) on each declared nuclear site, and upon request of the IAEA, identified locations outside a site which the IAEA considers might be functionally related to the activities of that site[49]
- Locations engaged in the activities relating to the manufacture and export of sensitive nuclear-related technologies[50]
- Exports, and upon a request by the IAEA, imports of specified equipment and non-nuclear material[51]
- Uranium mines and concentration plants, as well as thorium concentration plants
- Source material which has not reached the starting point of safeguards as defined in INFCIRC/153 (Corrected)

[48] Nuclear fuel cycle-related research and development activities are defined in Article 18 of the AP.
[49] Site is defined in Article 18 of the AP.
[50] The list of activities is specified in Annex I of the AP, and may be amended from time to time upon agreement by the Board of Governors.
[51] The list of specified equipment and non-nuclear material is specified in Annex I of the AP, and may be amended from time to time upon agreement by the Board of Governors.

- Exempted nuclear material
- Processing and storage of high or intermediate level waste containing plutonium, highly enriched uranium or uranium-233 material for which safeguards have been terminated
- Long-term plans for the State's nuclear fuel cycle development (covering a 10-year period).

Whereas Article 2 defines what is to be provided, Article 3 delineates when the information is to be provided, both initially and on a routine update basis.[52] Depending on the specific Article 2 provision, the relevant update needs to be transmitted (in accordance with Article 3) either quarterly, annually, as requested by the IAEA, or as agreed with the IAEA. There are the following important practical considerations a 'newcomer' State may want to keep in mind regarding the provision of information.

- Factors such as the site complexity, regulatory responsibilities, leasing arrangements that include tenants that may or may not be related to the nuclear fuel cycle (e.g., private companies), the temporary existence of construction tents, level of descriptive information to provide, could all add to the complications of preparing the requisite AP declaration. Early discussions with the IAEA, and perhaps other SSACs, are likely to make the process less daunting.
- The IAEA developed guidelines in 1997 to assist in the preparation and formatting of the AP declaration. Since then, the guidelines (IAEA, 2004) have been updated and reissued, and should be considered as an additional source of helpful information.
- It should not be considered unusual if the IAEA has a need for amplification or clarification of information submitted by the State pursuant to the AP. If such an occasion arises, the responsible National Authority[53] will be informed accordingly, and as experience shows, often the issue can be addressed through timely communication between the State and IAEA.
- It is a relatively more serious safeguards matter when a question or inconsistency is brought to the attention of a State. In such cases, it is to the benefit of all parties to resolve the issue(s) in a timely manner

[52] A State with no nuclear facilities and no nuclear material will still need to send an AP declaration pursuant to each Article 2 provision, which should state 'None' if there is nothing to report. A State which has submitted a declaration previously, and must now submit an update to that declaration pursuant to Article 3, should specify either 'No change' (if there are no changes from the previously reported declaration(s)), or only the changes that need be reported in the updated declaration.

[53] It is the State's responsibility to designate a responsible National Authority for implementing the measures prescribed by the relevant AP. Often this is SSAC for practical reasons, though it is not a requirement.

through close consultation and good cooperation between the State and the IAEA.

- Computer software[54] to help States with the submission of their AP declarations is available from the IAEA.

Complementary access (CA) is a measure which complements the access rights in the relevant Safeguards Agreement by provisioning the right of the IAEA to go to certain additional locations in a State for specific reasons as provided for in the AP. Complementary access is not an inspection, nor is it a right for the IAEA to go anywhere in a State for any reason whatsoever. Its implementation is exercised by the IAEA in accordance with the relevant articles of the AP.

With reference to the provisions of Articles 5 and 9 of the Model Additional Protocol (INFCIRC/540 (Corrected)) (IAEA, 1997a), the IAEA has a right to access all places on the declared sites of facilities and locations outside facilities, all other places where nuclear material is declared located, decommissioned facilities and LOFs, locations declared by the State where other nuclear fuel cycle-related activities are conducted, and other locations (under certain circumstances).[55] When a complementary access is to be performed by the IAEA, it must always be carried out in an objective and impartial manner. Normally, it is initiated via written correspondence from the IAEA to the State.[56] The advance notification to the State specifies the location(s) to be accessed, along with the reasons for access and the activities to be carried out during such access. The list of the authorized activities to be performed depends on the location to be accessed. Examples of such activities, as reflected in Article 6 of the Model AP, include visual observation; collection of environmental samples; use of radiation detection and measurement devices; examination of safeguards relevant production and shipping records; examination of records relevant to the quantities, origin and disposition of material; and/or placement of seals and other identifying and tamper-indicating devices specified in Subsidiary Arrangements.[57]

[54] The current version of the IAEA software is referred to as Protocol Reporter 2, development of which was completed in 2008.

[55] This includes locations of nuclear fuel cycle research and development not involving nuclear material; locations declared as manufacturing items listed in Annex I of the AP; locations declared as receiving imports of items listed in Annex II of the AP; and locations outside a site that the IAEA considers might be functionally related to a declared site.

[56] Under Article 8 of the AP, the State may offer access to locations in addition to those referred to in Articles 5 and 9, or it may request the Agency to conduct verification activities at a particular location.

[57] Other objective measures may be authorized under prescribed conditions, such as if demonstrated to be technically feasible, their use has been agreed by the Board of Governors and/or after consultation between the IAEA and the State.

Experience shows that a complementary access can be conducted efficiently and effectively the more technically capable the National Authority representative is (regarding the use of IAEA authorized equipment, collection of environmental samples, and other safeguards measures permitted under the AP). At certain times, for reasons relating to the sensitivity of information,[58] a State seeks to manage access to selected equipment, technology and processes. In such cases, the State may request 'managed access' provision in accordance with Article 7 of the protocol. These considerations are a normal part of the conduct of a complementary access, and the IAEA inspectors are trained to consult with the designated National Authority and the operator to find appropriate alternative methods or options to achieve the objectives of the complementary access.

Depending on the needs of the State concerned, the IAEA may be able to offer other services and technical assistance, such as AP-specific training or regional workshops on safeguards implementation. In addition, several countries (such as Australia, Japan and the United States) have in the past supported outreach programmes for developing countries, including sponsorship of regional or international seminars and training workshops, which may serve to enhance a State's technical capability and/or readiness for implementing the AP and its related safeguards commitments. Contact through the respective government mission to the IAEA or directly through the IAEA will often provide an understanding of such possibilities.

One benefit arising from the implementation of the AP is that integrated safeguards (IS) can be implemented in a State where the IAEA has been able to draw a broader safeguards conclusion for the State.[59] IS refers to an optimized combination of all safeguards measures available to the IAEA under the CSA and AP, to maximize effectiveness and efficiency within available resources.[60] With the increased assurances of the absence of undeclared nuclear material and activities in the State as a whole, implementation of IS takes into account a reduction in the traditional level of safeguards verification effort expended on less sensitive nuclear material (e.g., low enriched, natural and depleted uranium and irradiated fuel). For States with

[58] The State may make arrangements (referred to as 'managed access') with the IAEA to prevent the dissemination of proliferation sensitive information, to meet safety or physical protection requirements, or to protect proprietary or commercially sensitive information.

[59] As part of the State evaluation process, once the IAEA finds no indication of the diversion of declared nuclear material from peaceful nuclear activities and no indication of undeclared nuclear material or activities, it may issue a 'Broader Conclusion' that all nuclear material remained in peaceful activities.

[60] See IAEA publication *The Safeguards System of the International Atomic Energy Agency*, Para. 49, p. 14, available from http://www.iaea.org/OurWork/SV/Safeguards/safeg_system.pdf

an expanded nuclear fuel cycle, this has been shown to have a significant impact on resource utilization (i.e., shifting the focus to nuclear facilities, activities and material with higher strategic value). The same benefits are available to countries embarking on an expanded nuclear power programme; therefore, incorporation of the AP as part of a country's safeguards obligation should be given due consideration, if it is not already in force.

Should a NNWS decide positively on the inclusion of the AP as part of its safeguards agreement with the IAEA, the IAEA should be consulted early for the provision of advice and support to place the protocol in force. During this process, the State will need to determine which National Authority should have the responsibility for assuring that the objectives of the AP are fully achieved. As part of its options, it may consider adding the AP-related responsibilities to those of its designated SSAC. It is not a requirement, but many countries find this beneficial to both the State and the IAEA. In any case, examples of the type of AP-related responsibilities given to the National Authority include:

- Coordination and preparation for the submittal of relevant AP declarations
- Verification of declarations for correctness and completeness
- Processing and transmittal of declarations to the IAEA
- Responding to IAEA requests for additional information or clarification
- Facilitating the conduct of complementary access
- Resolving any AP-related questions and inconsistencies.

13.4.3 State system of accounting for and control of nuclear material

Milestones 1–3

Conceptually, in a safeguards regime, the national objective of the SSAC is to account for and control all nuclear material in the State. The international objective of the SSAC is to provide the essential basis for the application of IAEA safeguards and to support relevant regional or bilateral safeguards (such as those relating to nuclear-weapons-free zone treaties). In practice, many SSACs aim to meet both the national and international objectives for nuclear material accounting and control, plus the international objective of complying with other safeguards obligations (e.g., AP if in force, relevant nuclear-weapon-free zone treaty commitments, bilateral agreements with other States). In meeting these objectives, the main features of an SSAC would generally include at a minimum:

- The National Authority designated as the SSAC is independent of the facility operators.
- A framework is established (including legal and organizational elements) that codifies the SSAC's areas of responsibility, authority and regulation/control.
- Organizational and functional elements to support the safeguards regime are in place at the State level.
- Organizational and operational elements to support the safeguards regime are in place at the facility level as applicable.

From an operational point of view, the functions of an SSAC are carried out at two levels: the State level, implemented by the National Authority designated as the SSAC, and the facility level, which is implemented by facility operators. The State level is often responsible for the establishment of performance standards and implementation of safeguards requirements, and secondly, for confirming that the standards are maintained and the requirements met. An example of a performance standard is that the state system of accounting for and control of nuclear material[61] is effective concerning:

1. Maintaining and processing records of all nuclear material (showing types, amounts, locations, transfers) and of responsible individuals
2. Evaluating and reviewing the operation of the system for loss mechanisms, shipper/receiver differences, Material Unaccounted For (MUF), and measurement uncertainties associated with MUF.

An example of a safeguards requirement, by contrast, is that the SSAC is able to:

1. Support and maintain records of IAEA activities in the State (inspections and complementary access)
2. Handle the information required by regional or bilateral safeguards agreements with other States
3. Prepare reports and declarations for internal evaluation and for submission to outside bodies (e.g., the IAEA, other States and the Government), including AP declarations required by Articles 2 and 3 (if the AP is in force)[62]

[61] In this context, SSAC refers to the nuclear material accounting and control system (including the relevant hardware, software, procedures, policies and practices), and not to the National Authority which is also referred to as the SSAC.

[62] The AP requires some reporting on additional nuclear material (e.g., source or exempted material). There is no requirement for a State to designate the SSAC as entity responsible for the additional protocol implementation; experience shows however, there is an advantage for a State to extend the AP-related responsibilities to the same National Authority as the SSAC as it is experienced in working with the IAEA.

4. Assemble the safeguards relevant information together, facilitate analysis, and record findings
5. Report to the Government.

To confirm that the standards are maintained and the requirements met, the SSAC may consider the conduct of an audit and verification programme with the following objectives:

1. Verifying the correctness and completeness of submitted accounting and operating records and evaluating data for abnormal trends
2. Examining the facility design information and available/proposed operating practices presented in the nuclear facility licence/permit application to see if relevant safeguards objectives can be met
3. Ensuring the capability and performance of operators to account for and control nuclear material as required by both the State and the IAEA
4. Ensuring the accounting and control measures are adequate and effective to conclude the absence of unauthorized removal or use of the nuclear material
5. Conducting inspections during construction, commissioning and start-up of a facility, to confirm that the approved nuclear material accounting and control arrangements have been implemented
6. Performing audits investigating the qualification and training of key personnel
7. Performing audits investigating the accuracy of nuclear material measurement systems
8. Verifying the correctness and completeness of submitted additional protocol declarations (if applicable).

At the facility level, the operator is confronted with implementing the relevant safeguards requirements contained in the CSA and AP (if in force), in addition to its other requirements mandated by the State (e.g., safety, security, radiation protection in accordance with the facility operating licence and other requisite permits/licences at the national, regional or local level). From a safeguards perspective, this involves meeting or exceeding the safeguards-relevant standards and performance requirements laid down by the SSAC. In this respect, some of the most important safeguards-related functions to be performed by an operator involve:

1. Maintaining a system of nuclear material accounting and control, and reporting of nuclear material accounting and operating records
2. Maintaining and reporting safeguards-relevant facility design information (including, in cases of planned facilities, the early provision of design information) and facility design changes
3. Preparing and reporting additional protocol-relevant information, if applicable

4. Responding to IAEA or SSAC requests for clarification or explanation
5. Provisioning IAEA access to appropriate locations for the conduct of inspections, design information verification visits, and where applicable, complementary access, or for purposes related to the application of IAEA containment and surveillance systems (e.g., installation, maintenance, servicing, removal)
6. Addressing questions or inconsistencies identified by the IAEA or SSAC
7. Resolving any open discrepancy or anomaly if applicable.

13.4.4 Developing an effective SSAC

Milestone 1

In the early stages of development of a State's nuclear power programme, valuable advice on staffing levels and organizational structure/responsibilities may be realized by asking other SSACs on their experience and perspectives. In doing so, the questioner will be exposed to various organizational concepts and experiences, and at the same time, will see there are a variety of views on other State responsibilities that may be assigned to the SSAC staff (e.g., responsibility for safety, security, import/export control, safeguards training). Such input is also helpful when assessing what may be needed in order to increase the effectiveness of an established SSAC, and determining how best to tailor the responsibilities and staffing level of the SSAC to the individual needs of the host country.

The size of an SSAC organization is ultimately determined by the responsibilities assigned to it, financial considerations and the experience level and effectiveness of the staff members in carrying out their assigned responsibilities. An SSAC may require only one or two professional staff in the beginning, assuming only a CSA is in force, there are no nuclear facilities, and there are no or only small quantities of safeguarded nuclear materials in the State. As the nuclear programme in the State is developed further, such as when the first nuclear power plant (or research reactor) is under construction or an AP is to be brought into force, the State would want to start looking ahead as the required technical capacity/capability of the SSAC will need to grow in size and importance.

Milestones 2 and 3

Naturally, the more developed a State's nuclear fuel cycle becomes, the greater is the need for personnel resources in the SSAC. In this respect, consider a State with both a CSA and an AP in force, with nuclear fuel to be delivered to the State's first nuclear power plant and reportable but limited nuclear research and development in the country. In such a situation, the SSAC may find that three to five dedicated staff [63] is the minimum necessary to feel well positioned to address all safeguards-relevant requirements arising from the CSA and AP.

Other IAEA experience has demonstrated that a properly equipped, technically competent and capable SSAC can positively contribute to the IAEA's ability to confirm that a State's use of nuclear energy is exclusively for peaceful purposes. It is part of the reason why the IAEA sponsors regional SSAC training courses and offers various forms of technical assistance and other services associated with the application of safeguards. Two of these IAEA advisory services, for example, are the Integrated Nuclear Infrastructure Review (INIR) and the IAEA SSAC Advisory Service (ISSAS).

The INIR focuses on a holistic overview of national infrastructure development within a State with the objective to review the overall status of the development of the national nuclear power programme.[64] In advance of this mission, the IAEA also offers a Milestones mission (to provide an overview of an integrated approach to nuclear power planning) and self-assessment support (to assist a State with the process for conducting a self-assessment as well as how to understand the basis of evidence for each of the 19 issues in the evaluation methodology provided in IAEA Milestones document (IAEA, 2007a)). In any case, it may be reassuring to the 'newcomer' that the IAEA will undertake to explain the IAEA guidance publications and available services, and discuss future actions the State may wish to take.

The ISSAS[65] mission provides a requesting Member State, through the relevant National Authority, with recommendations and suggestions for improvements to its SSAC. These IAEA missions seek to evaluate the regulatory, legislative, administrative and technical components of the SSAC at both the State and facility level, and assess how the SSAC meets the obligations contained in the State's safeguards agreement and AP as applicable.

In addition to the above advisory services, periodically the IAEA sponsors international SSAC training courses to further develop the technical

[63] Consideration should be given to the higher range when the safeguards-experience level of the staff is limited, as these staff will require more time training and may be less effective in developing and implementing the safeguards policies and procedures compared to their more experienced colleagues.

[64] An IAEA brochure giving guidelines for the team leader and team members conducting an INIR is available from http://www-pub.iaea.org/MTCD/publications/PDF/INIR_Booklet.pdf

[65] Such a mission is conducted using the ISSAS Guidelines – Services Series 13, IAEA, Vienna, November 2005, available from http://www-pub.iaea.org/MTCD/publications/PDF/svs_013_web.pdf

capability and effectiveness of staff in the national authorities and facility operator(s) with the role and responsibility for the SSAC functions. Further, some Member States to the IAEA also provide technical support and other services to enhance an SSAC capability and performance. A key point here is that depending on a State's self-assessed needs, a State undertaking to build a nuclear power programme may find it beneficial to have a team of outside experts review the existing infrastructure, and offer safeguards-relevant recommendations regarding such complex issues as:

- Requirements for nuclear material accounting and control (including international transfers)
- Conditions for possession of nuclear material (e.g., who can own, transfer, and/or use nuclear material, and under what conditions)
- Requirements for prompt notification in the event of losses, unauthorized use, removal of nuclear material, and responses to them
- Requirements for submission of CSA and AP declarations
- Legal authority for the SSAC to impose its requirements
- Requirements for granting access to State and IAEA inspectors.

Globally, regionally and at the national level, increased safeguards effectiveness may also arise through other State-initiated actions as well. In this regard, a State may wish to consider, in the context of its national interests:

- Assigning functional responsibility to the SSAC as the point of contact for regional or bilateral safeguards matters with other regional/international organizations
- Integration of Safety, Security and Safeguards (3S) functions and responsibilities (Kovacic *et al*. (2009) have developed a recommended starting point for discussion on the 3S concept)
- Assessing whether there are actionable steps that may be taken to maximize the non-proliferation benefit arising from Article VIII.A of the Statute of the IAEA which specifies that 'each member should make available such information as would, in the judgment of the member, be helpful to the Agency'
- Participation in the IAEA's Member State Support Program consistent with national capability and resources
- Participation in and/or sponsorship of safeguards-relevant training at the international, regional and national levels consistent with national capability and resources
- Provision of safeguards technical assistance to responsible facility operators and other stakeholders in the State.

At the same time, the SSAC and facility operators also have much to gain by consulting early with the IAEA. For example, early consultations are shown to have a positive impact on the ability of the SSAC and nuclear facility operators to work effectively with the IAEA inspectors to verify an initial inventory of declared nuclear material, or they have helped to ensure that the SSAC provides timely, correct and complete State declarations to the IAEA, or they have enhanced the facility operator's ability to facilitate timely access and support during inspections, facility design information verification visits or complementary accesses.

At this juncture, it may be self-evident that the SSAC will be actively and routinely working towards fulfilment of the State's safeguards-relevant obligations as part of achieving Milestones 2 and 3, including for example, the provision of the requisite information to the IAEA, facilitating IAEA access to facilities and LOFs for the purpose of conducting inspections and design information verification visits, auditing the facility operator's conduct of annual physical inventory takings, and performing other safeguards-relevant activities (e.g., responding to IAEA requests for the conduct of complementary access if applicable). In any case, an important point for a 'newcomer' to remember is that increased SSAC effectiveness is acquired through a high level of technical competence of the SSAC in an environment of good cooperation between the SSAC, facility operators and the IAEA.

13.4.5 Other international non-proliferation obligations

Milestone 1

As reflected by the IAEA, no safeguards system, no matter how extensive the measures put in place, can provide absolute assurance that there has been no diversion of nuclear materials or that there are no undeclared nuclear material or activities in a State. The safeguards system for implementing comprehensive safeguards agreements, including additional protocols, is designed to provide for verification by the IAEA of the correctness and completeness of States' declarations, so that there is credible assurance of both the non-diversion of declared nuclear material from peaceful activities and the absence of undeclared nuclear material and activities.

Further to a State's undertakings, Article III of the NPT also stipulates specific obligations of each State party to the NPT 'not to provide: (a) source or special fissionable material, or (b) equipment or material especially designed or prepared for the processing, use or production of special fissionable material, to any non-nuclear-weapon State for peaceful purposes, unless the source or special fissionable material shall be subject to the safeguards required by this article.' After the entry into force of the NPT, multilateral consultations on nuclear export controls led to the establishment of two separate forums for dealing with nuclear exports: the

Zangger Committee in 1971 and the Nuclear Suppliers Group (NSG) in 1975.[66]

The Zangger Committee[67] was set up to consider procedures for exports of nuclear material and equipment related to NPT commitments. In August 1974, the committee produced a trigger list of items which would require the application of IAEA safeguards and, if the items were to be exported directly or indirectly to a NNWS which was not party to the NPT, the application of export procedures by the supplier. The trigger list and associated guidelines were communicated to and published by the IAEA as INFCIRC/209 (IAEA, 1974), which has been updated several times.

The Nuclear Suppliers Group, also known as the London Group or London Suppliers Group, was set up in 1974 after India exploded its first nuclear device, and included both non-members and members of the Zangger Committee. An authoritative document on the history, role and activities of the NSG has been published by the IAEA as INFCIRC/539 as amended (IAEA, 1997c, 2000, 2003, 2005c, 2009). The group sought to ensure that transfers of nuclear material or equipment would not be diverted to unsafeguarded nuclear fuel cycle or nuclear explosive activities. Therefore, among the other conditions of supply in the NSG guidelines, formal government assurances to this effect were required from recipients. The guidelines were originally communicated to the IAEA in 1978 and published as INFCIRC/254 (IAEA, 1978); the guidelines have been periodically amended, including the addition of Part 1 (IAEA 1992a) and Part 2 (IAEA, 1992b).

Milestones 2 and 3

In a practical sense, the NSG guidelines (INFCIRC/254, Parts 1 and 2) are essentially a set of export rules which govern the export of items and technologies especially designed or prepared for nuclear use (Part 1) and the export of nuclear-related dual-use items and technologies (Part 2).[68] Whereas States party to the NPT have already forsworn the acquisition of nuclear weapons and other nuclear explosive devices, and agreed to accept comprehensive safeguards on the entirety of their nuclear fuel cycle, States

[66] For authoritative information on the NPT Review Conferences, the Zangger Committee and the Nuclear Suppliers Group through 1995, see NPT/CONF.1995/7/Part II, 18 April 1995, available from http://www.un.org/depts/ddar/nptconf/2136.htm

[67] Also known as the 'NPT Exporters Committee'.

[68] The guidelines effectively require nuclear suppliers to exercise special care in the export of sensitive facilities, technology and weapons-usable material, for example reprocessing and isotope separation facilities.

not signatory to the NPT will need to consider the nuclear suppliers' requirements for export of sensitive facilities, equipment, material and technologies used for peaceful nuclear purposes, prior to deciding whether to construct a nuclear facility.

To control the non-proliferation of nuclear material and technologies, the international community has focused on both States and non-State actors. Some requirements and measures extend beyond nuclear non-proliferation to non-proliferation of weapons of mass destruction and are included in several legally binding instruments by the UN Security Council (UNSC). Of primary importance is United Nations Security Council Resolution 1540 (UNSCR 1540),[69] adopted in April 2004. UNSCR 1540 (2004) requires that all States adopt and enforce appropriate laws that prohibit any non-State actor to manufacture, acquire, possess, develop, transport, transfer or use nuclear, chemical or biological weapons (otherwise known as weapons of mass destruction or WMD) and their means of delivery. Among its provisions is the call upon all States to fulfil their commitment to multilateral cooperation, including those within the framework of the IAEA.

A 'newcomer' will inevitably benefit from understanding the scope of UNSCR 1540 (2004)[70] and related resolutions (e.g., UNSCR 1673 (2006) and UNSCR 1810 (2008))[71] regarding their implications at the State level, to establish national control over WMD-related material in the areas of accounting/securing, physical protection, border and law enforcement, export and transshipment, for example. However, as the objectives of these resolutions go well beyond safeguards, they are outside the scope of this chapter. Nevertheless, these UNSC resolutions, and other resolutions with their relevant measures to be implemented accordingly, are part of the legally binding instruments and commitments every State is obligated to undertake. They should be factored into the information acquisition process inherent to the IAEA's Milestones 1–3 as discussed in the Milestones publication (IAEA, 2007a).

[69] UNSCR 1540 (2004) is available from the 1540 Committee web page at http://www.un.org/sc/1540/

[70] The UN Security Council has published a 'Frequently Asked Questions on UNSCR 1540' web page, available from http://www.un.org/sc/1540/faq.shtml#1

[71] The Security Council extended the mandate of the 1540 Committee for a further two years with the adoption of Resolution 1673 (2006), which reiterated the objectives of Resolution 1540 (2004), expressed the interest of the Security Council in intensifying its efforts to promote full implementation of the resolution, and obliged the 1540 Committee to report again by April 2008. Then in April 2008, the Security Council adopted Resolution 1810 (2008), which extended the mandate of the 1540 Committee for a further period of three years, thereby reaffirming the objectives of Resolution 1540 (2004) and Resolution 1673 (2006).

13.5 Transparency during a nuclear renaissance

Safeguards are an essential part of international confidence-building measures, and serve to help demonstrate a country's commitment to non-proliferation. But today's safeguards will likely need to adapt to tomorrow's challenges and with those challenges will come new incentives for countries to become more transparent. In preparing for tomorrow's challenges, the IAEA has considered the future environment, and reported its internal assessment in February 2008, in '20/20 Vision for the Future, Background Report by the Director General for the Commission of Eminent Persons'.[72] This report presents the results of a review by the IAEA regarding the role of the IAEA through the year 2020 and beyond. While the publication will benefit a 'newcomer' on the potential future direction of the IAEA and safeguards, its 'foresight' analysis and forward-looking review may be of particular interest to those stakeholders who wish to consider the longer term in terms of transparency and the non-proliferation regime. For example, in the Executive Summary, it says:

> Although a revival in nuclear power would require additional verification ('safeguards') activities, the IAEA's workload is not likely to increase proportionally if States accept greater transparency measures under a new verification standard. The need for IAEA inspectors in the field is likely to decrease due to the use of new technology and a change in the way States are evaluated. Verification activities will increasingly become information driven, with more evaluation work at the Agency's headquarters. Meeting future challenges will require a robust IAEA 'toolbox' containing the necessary legal authority to gather information and carry out inspections, state-of-the-art technology, a high calibre workforce and sufficient resources.

For stakeholders, one pertinent question raised by the above statement is: what will States accept as greater transparency measures under a new verification standard?

Demands for greater transparency about another State's nuclear activities arise for a variety of reasons, including the desire of States to understand the nuclear capabilities and policies of other States. Berkhout and Walker (1999) have considered this question. In terms of transparency mechanisms applied during the development of a nuclear power infrastructure, one should keep in mind that there are the expressed and implied needs of stakeholders at the international, regional, national, sub-national and local levels which should be considered as part of the decision process (i.e., prior to making the decision to develop a nuclear power infrastruc-

[72] Available from http://www.iaea.org/NewsCenter/News/PDF/20-20vision_220208.pdf

ture). At the same time, while implementation of transparency mechanisms (and other confidence-building measures) clearly will have benefits at each of these levels, the potential for negative impacts must also be explicitly addressed as reported by Harmon *et al.* (2000) from the Sandia National Laboratories in a report for the US Department of Energy. This is particularly important in view of the fact that a major reason justifying secrecy is non-proliferation. The outcome of such an analysis (whether formal or informal, whether part of a broader analysis of national security objectives, or narrowly defined at the facility level), will enable a State to better align its national interests with its non-proliferation objectives.

And what is the appropriate level of nuclear transparency? That is a question for which each stakeholder forms his or her own opinion. Some suggest that one example of the appropriate level of transparency is illustrated by the transparency mechanism applied to the exclusively peaceful uses of nuclear energy between two States, namely Argentina and Brazil, as discussed by Fernandez-Moreno and D'Amato (2002) in the 24th Annual Meeting of the European Safeguards Research and Development Association, ESARDA. Johnston *et al*. (2008) consider that 'the point and the measure of transparency is full and open truthfulness while being mindful that complete transparency is an abstraction that will never be fully achieved in any society'. Others may differ. Fortunately, transparency in the nuclear field and its contribution to non-proliferation continue to be discussed in several international forums, more recently in the context of the 2010 NPT Review Process that was referred to in Section 13.2.1, Birth of a landmark treaty.

Noting that IAEA safeguards agreements are in force in every State thought to have nuclear activities, but recalling that some States have yet to conclude a safeguards agreement[73] with the IAEA as required by the NPT, transparency remains a subject of global interest (and one that is open to differing points of view). As recognized by the IAEA, an expansion of nuclear power will call for ever greater transparency.[74] A State's ability to fully embrace and adhere to its international obligations arising from the NPT can well serve as a foundation for building transparency in an age of nuclear renaissance. Besides concluding a safeguards agreement with the IAEA (if none is in force), an example of how a State may increase transparency is found in the statement of the IAEA Director General Yukiya

[73] Status of IAEA safeguards agreements, quantities of nuclear material and facilities safeguarded, and other relevant information for each State is available from http://www.iaea.org/OurWork/SV/Safeguards/sv.html

[74] IAEA, 20/20 Vision for the Future, Background Report by the Director General for the Commission of Eminent Persons, p. 16, February 2008, available from http://www.iaea.org/NewsCenter/News/PDF/20-20vision_220208.pdf

Amano in March 2010, concerning international cooperation being vital to the nuclear renaissance:[75]

> Responsibility means countries must abide by the highest safety and security standards and implement IAEA safeguards so the Agency can verify that nuclear materials are being used exclusively for peaceful purposes. All countries with nuclear power should adhere to the Convention on Nuclear Safety and the Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management. All countries are encouraged to implement a so-called *Additional Protocol* to their safeguards agreement with the IAEA, which boosts transparency by giving the Agency's inspectors more authority.

These views are just some of the many ideas and concepts involving transparency, and they should be factored in as part of a State's progression in the development of its nuclear power infrastructure. In doing so, the concerned State may be better positioned to advance its national interest and achieve its non-proliferation goals and objectives.

13.6 Sources of further information and advice

Regarding further information on non-proliferation and safeguards, a true list of web sources would be exhaustive, and in today's rapidly changing web environment that list might very well be out of date the moment it is published. To be of service to the 'newcomer', the sources listed below are essentially limited to UN and IAEA web pages, which many people would consider to be authoritative. Nevertheless, there are many other sites available, and as the 'newcomer' progresses in their search, he or she will undoubtedly uncover a host of these other websites, many of which are associated with both NGOs and governmental organizations involved with and/or responsible for non-proliferation and safeguards.

13.6.1 Web-based general sources related to the NPT

Information related to the NPT, with links to associated international safeguards, is provided at the IAEA's web pages located at:

- http://www.iaea.org/Publications/Documents/Treaties/npt.html
- http://www.iaea.org/Publications/Documents/Treaties/index.html
- http://www.iaea.org/OurWork/SV/Safeguards/legal.html
- http://www.iaea.org/OurWork/SV/Safeguards/sv.html

[75] Director General Amano's statement as published in *Le Monde*, OpEd, 7 March 2010; full statement available from http://www.iaea.org/newscenter/transcripts/2010/lm070310.html

For those readers who may want to become more familiar with the developments of the NPT (both historical and present day), a recommended starting point is the 'NPT Briefing Book (MCIS/CNS) 2010', available from http://cns.miis.edu/treaty_npt/npt_briefing_book_2010/index.htm.

13.6.2 Web-based general sources on non-proliferation and disarmament

- http://www.iaea.org/Publications
- http://unhq-appspub-01.un.org/UNODA/TreatyStatus.nsf
- http://www.opanal.org/NWFZ/nwfz.htm
- http://www.un.org/disarmament
- http://www.unidir.org/
- http://www.unog.ch/disarmament

13.6.3 Web-based safeguards-relevant sources and publications

- http://www.iaea.org/Publications/Magazines/Bulletin/Bull511/51103570609.html
- http://www.iaea.org/OurWork/SV/Safeguards/safeg_system.pdf
- http://www.iaea.org/Publications/Booklets/Safeguards3/safeguards0408.pdf
- http://www.iaea.org/Publications/Booklets/Safeguards3/safeguards0707.pdf
- http://www.iaea.org/Publications/Booklets/Safeguards3/safeguards0806.pdf
- http://www-pub.iaea.org/MTCD/publications/PDF/NVS1–2003_web.pdf

13.7 References

Amano (2010), Beneficial, responsible, sustainable, *IAEA Bulletin*, 51(2), April 2010. Available from http://www.iaea.org/Publications/Magazines/Bulletin/Bull512/51204711415.pdf

Berkhout F and Walker W (1999), *Transparency and Fissile Materials, Fissile Materials: Scope, Stocks and Verification,* Disarmament Forum, UNIDIR, Geneva. Available from http://www.fas.org/nuke/control/fmct/2e-berkh.pdf

Fernandez-Moreno S and D'Amato E (2002), *Establishing Confidence Building Measures Between Countries in the Nuclear Field*, 24th Annual Meeting of ESARDA, Workshop on Research and Development Responses to New Safeguards Environment, Luxembourg. Available from http://200.0.198.11/MenoriaT/MT-02/MT16-02.pdf

Fischer D (1997), *History of the IAEA: The First Forty Years,* A Fortieth Anniversary Publication. IAEA, Vienna. Available from http://www-pub.iaea.org/MTCD/publications/PDF/Pub1032_web.pdf

Fischer D (2003), Vision & Reality: How far has the IAEA been able to realize the vision that inspired its creation in 1957?, *IAEA Bulletin*, 4/52. Available from http://www.iaea.org/Publications/Magazines/Bulletin/Bull452/article4.pdf

Harmon CD, Olsen JN and Passell HD (2000), *Nuclear Facility Transparency: Definitions and Concepts*, SAND2000–24000C, Sandia National Laboratories, Albuquerque, NM. Available from http://www.osti.gov/bridge/servlets/purl/764841-aaZdAG/webviewable/

IAEA (1959), *The Texts of the Agency's Agreements with the United Nations*, INFCIRC/11, IAEA,Vienna. Available from http://www.iaea.org/Publications/Documents/Infcircs/Others/infcirc11.pdf

IAEA (1968), *The Agency's Safeguards System (1965, as Provisionally Extended in 1966 and 1968),* INFCIRC/66/Rev.2, IAEA, Vienna. Available from http://www.iaea.org/Publications/Documents/Infcircs/Others/infcirc66r2.pdf

IAEA (1970), *Treaty on the Non-Proliferation of Nuclear Weapons*, INFCIRC/140, IAEA, Vienna. Available from http://www.iaea.org/Publications/Documents/Infcircs/Others/infcirc140.pdf

IAEA (1972), *The Structure and Content of Agreements between the Agency and States Required in Connection with the Treaty on the Non-Proliferation of Nuclear Weapons*, INFCIRC/153 (Corrected), IAEA, Vienna. Available from http://www.iaea.org/Publications/Documents/Infcircs/Others/infcirc153.pdf

IAEA (1974), *Communication Received from Members Regarding the Export of Nuclear Material and of Certain Categories of Equipment and Other Material*, INFCIRC/209, 3 September 1974, IAEA, Vienna. Original text and amendments available from http://www.iaea.org/Publications/Documents/Infcircs/Numbers/nr201-250.shtml

IAEA (1978), *Communications Received from Certain Member States Regarding Guidelines for the Export of Nuclear Material, Equipment and Technology*, INFCIRC/254 (amended), IAEA, Vienna. Available from http://www.iaea.org/Publications/Documents/Infcircs/Numbers/nr251-300.shtml

IAEA (1992a), *Communications Received from Member States Regarding Guidelines for the Export of Nuclear Material, Equipment and Technology Nuclear Transfers*, INFCIRC/254, Rev.1, Part 1, IAEA, Vienna. Available from http://www.iaea.org/Publications/Documents/Infcircs/Numbers/nr251-300.shtml

IAEA (1992b), *Communications Received from Member States Regarding Guidelines for the Export of Nuclear Material, Equipment and Technology Nuclear Transfers*, INFCIRC/254, Rev.1, Part 2, IAEA, Vienna. Available from http://www.iaea.org/Publications/Documents/Infcircs/Numbers/nr251-300.shtml

IAEA (1997a), *Model Protocol Additional to the Agreement(s) between State(s) and the International Atomic Energy Agency for the Application of Safeguards*, INCIRC/540 (Corrected), IAEA, Vienna. Available from http://www.iaea.org/Publications/Documents/Infcircs/1997/infcirc540c.pdf

IAEA (1997b), IAEA turns 40. Key dates and historical developments, Supplement to *IAEA Bulletin*, 2/93, IAEA, Vienna. Available from http://www.iaea.org/Publications/Magazines/Bulletin/Bull393/Chronology/chronology.pdf

IAEA (1997c), *Communication Received from the Permanent Mission of Australia on Behalf of Member States of the Nuclear Suppliers Group*, INFCIRC/539, IAEA, Vienna. Available from http://www.iaea.org/Publications/Documents/Infcircs/1997/inf539.shtml

IAEA (2000), *Communication Received from the Permanent Mission of Nederland on behalf of Member States of the Nuclear Suppliers Group*, INFCIRC/539 Rev. 1, IAEA, Vienna. Available from http://www.iaea.org/Publications/Documents/Infcircs/2000/infcirc539r1corrected.pdf

IAEA (2001), *IAEA Safeguards Glossary, 2001 Edition*. International Nuclear Verification Series No. 3, IAEA, Vienna. Available from http://www-pub.iaea.org/MTCD/publications/PDF/nvs-3-cd/PDF/NVS3_scr.pdf

IAEA (2003), *Communication Received from the Government of the United States of America on Behalf of Member States of the Nuclear Suppliers Group*, INFCIRC/539 Rev. 2, IAEA, Vienna. Available from http://www.iaea.org/Publications/Documents/Infcircs/2003/infcirc539r2.pdf

IAEA (2004), *Guidelines and Format for Preparation and Submission of Declarations Pursuant to Articles 2 and 3 of the Model Protocol Additional to Safeguards Agreement*, Service Series 11, IAEA, Vienna. Available from http://www-pub.iaea.org/MTCD/publications/PDF/svs_011_web.pdf

IAEA (2005a), *Appendix: How to Conclude a Comprehensive Safeguards Agreement with an Additional Protocol*, IAEA, May 2005, Non-Proliferation of Nuclear Weapons and Nuclear Security: IAEA Safeguards Agreements and Additional Protocols. Available from http://www.iaea.org/Publications/Booklets/nuke.pdf

IAEA (2005b), *Evolution of the Safeguards System 1999–2005*, Section C of Safeguards System of the IAEA, Department of Safeguards. IAEA, Vienna. Available from http://www.iaea.org/OurWork/SV/Safeguards/safeg_system.pdf

IAEA (2005c), *Communication Received from the Government of Sweden on Behalf of Member States of the Nuclear Suppliers Group*, INFCIRC/539 Rev. 3, IAEA, Vienna. Available from http://www.iaea.org/Publications/Documents/Infcircs/2005/infcirc539r3.pdf

IAEA (2007a), *Milestones in the Development of a National Infrastructure for Nuclear Power*, IAEA Nuclear Energy Series No. NG-G-3.1, IAEA, Vienna. Available from http://www-pub.iaea.org/MTCD/publications/PDF/Pub1305_web.pdf

IAEA (2007b), *Safeguards: Staying Ahead of the Game*, IAEA, Vienna. Available from http://www.iaea.org/Publications/Booklets/Safeguards3/safeguards0707.pdf

IAEA (2008a), *Evaluation of the Status of National Nuclear Infrastructure Development*, IAEA Nuclear Energy Series No. NG-T-32, IAEA, Vienna. Available from http://www-pub.iaea.org/MTCD/publications/PDF/Pub1358_web.pdf

IAEA (2008b), *Verifying Compliance with Nuclear Non-Proliferation Undertakings: IAEA Safeguards Agreements and Additional Protocols,* IAEA, Vienna. Available from http://www.iaea.org/Publications/Booklets/Safeguards3/safeguards0408.pdf

IAEA (2009), *Communication Received from the Resident Representative of Hungary on behalf of Member States of the Nuclear Suppliers Group*, INFCIRC/539 Rev. 4, IAEA, Vienna. Available from http://www.iaea.org/Publications/Documents/Infcircs/2009/infcirc539r4.pdf

Jennekens J (1970), IAEA safeguards: a look at 1979–1990 and future prospects, *IAEA Bulletin*, 3/21, IAEA, Vienna. Available from http://www.iaea.org/Publications/Magazines/Bulletin/Bull321/32103450410.pdf

Johnston RG, Maerli MB, Bitzer EG and Ballard JD (2008), Two simple models of nuclear transparency, *International Journal of Social Inquiry*, 1(2), pp. 201–235. Available from http://www.socialinquiry.org/articles/IJSI-V1N122008%20-%20009.pdf

Kovacic DN, Raffo-Caiado A, McClelland-Kerr J, Van Sickle M, Bissani M and Apt K (2009), *Nuclear Safeguards Infrastructure Development and Integration with Safety and Security*, ORNL OSTI ID: 975066, ORNL, Oak Ridge, TN. Available from http://info.ornl.gov/sites/publications/files/Pub20268.pdf

14

Spent fuel and radioactive waste management in nuclear power programmes

H. FORSSTRÖM, SKB International AB, Sweden

Abstract: The generation of spent nuclear fuel and radioactive waste is an unavoidable consequence of nuclear power production. Some of this material needs to be handled with great care and be disposed of, either near the surface for short-lived waste (a few hundred years) or at depth (500–1000 m) in geological formations for long-lived and high-level waste. The spent nuclear fuel also contains material (uranium and plutonium) that could be recycled in new fuel after reprocessing. This chapter provides an overview of the characteristics of spent fuel and different types of radioactive waste and of the steps involved in the management of this material. It also covers the international framework and national policies and strategies.

Key words: radioactive waste, spent nuclear fuel, reprocessing, disposal.

14.1 Introduction

The generation of radioactive waste is an unavoidable consequence of nuclear power production as well as of other applications of nuclear technologies, e.g. the use of radioactive substances in medicine or research. Some of the waste is very dangerous and needs to be handled with great care and be isolated from human beings and the environment. These wastes will also remain dangerous for very long time periods, from hundreds to hundreds of thousands of years. The end point of radioactive waste management is therefore in most cases disposal, either near the surface for short-lived waste (a few hundred years) or at depth (500–1000 m) in geological formations for the long-lived and high-level waste. To reduce the need for disposal one of the basic principles for radioactive waste management is to minimize the generation.

The main source of long-lived and high-level waste is the spent nuclear fuel. It also contains material (uranium and plutonium) that could be recycled in new nuclear fuel after reprocessing. The possible reuse will depend on the economic conditions and, in particular, the development of fast reactors. The alternative is to dispose of it directly after 30–40 years of storage.

The spent fuel is thus a good example of the following definition of waste: *Waste is a resource at the wrong time and in the wrong place*.

In addition to spent nuclear fuel, several other different types of secondary waste are generated during nuclear power production or during reprocessing and from the final decommissioning and dismantling of the reactors and auxiliary facilities. Most of this waste is short-lived and classified as low-level waste.

As the waste management will be applied over a hundred years or more it is important to develop appropriate policies and strategies for the waste management, including technical options and definition of clear responsibilities for regulating, implementing and financing the waste management system. Although many of the facilities (e.g. for disposal) will only be built several tens of years later, it is very important for a country considering the introduction of nuclear power to develop policies and strategies early.

The IAEA defines radioactive waste as any waste that contains or is contaminated with radionuclides at concentrations or activities greater than clearance levels as established by a regulatory body (IAEA, 2007a). It is recognized that this definition is purely for legal and regulatory purposes and that material with activity concentrations less than clearance levels is also radioactive from a physical point of view.

14.1.1 Sources of radioactivity

In a nuclear power plant there are two sources for the production of radioactive substances, the fission by neutrons and absorption of neutrons (transmutation by neutron absorption) taking place in the fuel itself, and the irradiation of material in the reactor that is exposed to the neutrons from the fission process (activation). The radioactive substances produced from the first source are fission products and transuranic elements (elements heavier than uranium). The fission products are the lighter elements (e.g. cesium, strontium and iodine) that are created when the heavier atoms (e.g. uranium or plutonium) are split (fissioned) and energy is released. The transuranic elements (e.g. plutonium, americium and curium) are generated by the absorption of neutrons in uranium and the successively created transuranic elements. The amount of fission products and transuranic elements is directly coupled to the energy that has been generated. The spent fuel is highly radioactive and will need shielding and cooling for the subsequent handling.

A typical composition of spent nuclear fuel is shown in Table 14.1. The fission products and transuranic elements are kept in the fuel and contained by the fuel cladding. They will only be released to other parts of a nuclear power plant if the fuel cladding is damaged. Minor amounts could also

Table 14.1 Composition of spent nuclear fuel (PWR, 60 MWd/kg U, 15 years out of the reactor): the most important radionuclides

Radionuclide	Half-life (years)	Activity (Bq/tU)	Radionuclide	Half-life (years)	Activity (Bq/tU)
Fission products			*Transuranic elements*		
H-3	12	2E+13	U-234	250,000	5E+10
Kr-85	11	2E+14	Np-239	2,100,000	3E+12
Sr-90	30	3E+15	Pu-238	88	3E+14
Y-90		3E+15	Pu-239	24,000	1E+13
Tc-99	210,000	8E+11	Pu-240	6,600	2E+13
Sn-121		5E+11	Pu-241	14	3E+15
Sn-121m	55	6E+11	Pu-242	370,000	2E+11
Sb-125	3	1E+13	Am-241	430	1E+14
Cs-134	2	6E+13	Am-242m	150	8E+11
Cs-137	30	5E+15	Am-242		8E+11
Ba-137m		4E+15	Am-243	7,400	3E+12
Pm-147	3	1E+14	Cm-242	0.4	7E+11
Sm-151	90	2E+13	Cm-243	29	1E+12
Eu-154	9	1E+14	Cm-244	18	3E+14
Eu-155	5	2E+13	Cm-245	8,500	7E+10
Total		2E+16	Total		4E+15

emanate from fuel contamination on the outside of the fuel cladding that remains after the fuel fabrication.

The second source of radioactive substances in a reactor, activation products, is the result of irradiation of material in the reactor by neutrons from the fission process. Only material inside the reactor pressure vessel and in the concrete that immediately surrounds it will be exposed to sufficient neutron fields for activation. The highest activity will be generated in the core components holding the fuel and in other internal parts in the pressure vessel. Also material contained in the coolant or coolant-moderator water, which passes through the reactor core, could become activated.[1] This could be metal ions or particles from corrosion in the primary circuit of the reactor or other trace elements contained in the coolant or coolant-moderator. Radioactive substances thus created could then be transported through the primary system of the reactor and contaminate surfaces and filters, thus creating a radiation field around these components and in the end a radioactive waste.

A list of typical activation products is given in Table 14.2. To minimize the creation of activation products, one strives to keep the primary circuit water very clean through ion exchange and mechanical filtering as well as

[1] In this chapter only light and heavy water reactors are considered.

Table 14.2 Activation products in fuel cladding and mechanical components (PWR, 60 MWd/kg U, 15 years out of the reactor): the most important radionuclides

Radionuclide	Half life (years)	Activity (Bq/tU)
Activation products		
C-14	5,700	6E+10
Fe-55	3	9E+12
Co-60	5	6E+12
Ni-59	75,000	1E+11
Ni-63	96	1E+13
Zr-93	1,500,000	1E+10
Nb-93m	14	2E+14
Nb-94	20,000	3E+11
Sn-121m	55	1E+11
Total		3E+14

to reduce the corrosion by adjusting the chemical environment, e.g. by adding lithium hydroxide or hydrazine to the coolant. Also gaseous radioactive fission and activation products are formed and transported by the coolant and coolant-moderator to a degasification system.

14.1.2 Classification of radioactive waste

Radioactive waste covers a wide spectrum of material types, physical composition and radioactivity concentration. Also the composition of radionuclides included in the waste and their corresponding half-lives differs widely. This means that the methods to take care of the radioactive waste will have to be adapted to the specific waste form. In particular it is important to distinguish between solid, liquid and gaseous waste, as well as to consider the radiation level at the waste package and the half-life of the radionuclides contained in the waste. The physical form of the waste (solid, liquid or gaseous) determines the treatment, conditioning and packaging methods to be used for the waste. The radiation level determines the handling and storage method for the waste, and the concentration and half-life of the radionuclides determines the way they need to be finally disposed of. Radionuclides with half-lives of 30 years or shorter are considered to be short-lived.

Earlier classification schemes distinguished between exempt waste, short-lived low- and intermediate-level waste, long lived low- and intermediate-level waste and high-level waste. Exempt waste had no radiological restrictions. Low-level waste could be handled without extra shielding, while intermediate and high-level waste required shielding for handling and high-level waste also required cooling.[2] Short-lived waste could be disposed of at or near the surface, while long-lived waste and high-level waste would require deep geological disposal. This and similar classification schemes are still being used in many countries.

More recently the IAEA has introduced a new classification scheme that is based on the way the waste will be finally disposed of (IAEA, 2009a). It has the following six classes of radioactive waste:

- Exempt waste (EW): Waste that meets the criteria for clearance, i.e. it has been cleared from regulatory control, and is not considered radioactive waste.
- Very short-lived waste (VSLW): Waste that can be stored for decay over a limited period of up to a few years and subsequently cleared for uncontrolled disposal, use or discharge.
- Very low-level waste (VLLW): Waste that does not necessarily meet the criteria of EW, but that does not need a high level of containment and isolation and, therefore, is suitable for disposal in near-surface landfill-type facilities with limited regulatory control.
- Low-level waste (LLW): Radioactive waste with only limited amounts of long-lived radionuclides. Such waste requires robust isolation and containment for periods of up to a few hundred years and is suitable for disposal in engineered near-surface facilities.
- Intermediate-level waste (ILW): Waste that, because of its content, particularly of long-lived radionuclides, requires disposal at greater depths, of the order of tens of metres to a few hundred metres.
- High-level waste (HLW): Waste with levels of activity concentration high enough to generate significant quantities of heat, or waste with large amounts of long-lived radionuclides. Disposal in deep, stable geological formations usually several hundred metres or more below the surface is the generally recognized option for disposal of HLW.

14.1.3 Radioactive waste from nuclear power production

Several kinds of radioactive waste are generated from nuclear power production. The most hazardous is the spent nuclear fuel (if considered as waste), or the high-level waste from chemical reprocessing of the fuel. Intermediate-level waste is mainly irradiated core components and some long-lived waste from reprocessing. Low-level waste comes from the treatment of the water in and off-gases from the reactor primary circuit and fuel

[2] The requirements on shielding and/or cooling are determined by the activity concentration in the waste. With a high activity concentration, in particular of α or β emitting radionuclides, much of the energy released through radiation is absorbed as heat in the material.

handling facilities and from components and material that have been in contact with such water or gases. Some of this waste could even qualify as very low-level waste. LLW and VLLW are generated both during the operation and maintenance of the nuclear power plants (and possible reprocessing plants) and during their final decommissioning and dismantling after power production has ceased. In particular, large volumes of VLLW are generated during dismantling.

Some minor amounts of radioactive substances are released from nuclear power plants during normal operation through the cooling water or with the off-gases. These amounts are strictly controlled and in compliance with regulatory limits. Such limits are set very low to ensure a very small radiological impact on the people and environment in the vicinity of the power plant. Different processes, e.g. filtration, ion exchange and evaporation, are used to minimize the releases. The normal operational releases from a power plant are not further dealt with in this chapter (see Chapters 11 and 17 for further details), which is dedicated to waste that will be further taken care of.

14.2 Policies and strategies for management of spent fuel and radioactive waste

14.2.1 Need for national policies and strategies

Spent fuel and radioactive waste will be generated from the first day of operation of a nuclear power plant. It needs to be taken care of through intermediate storage, treatment and conditioning, possible reprocessing and final disposal, steps that might very well stretch out over 100 years or more. Introduction of nuclear power involves a long-term commitment for the country and the industry involved. It is thus important to develop policies and strategies for their management, as well as a stable legal system, at an early stage of the decision process for implementing nuclear power in a country. The policies should include a general plan for the spent fuel and waste management systems needed and a clear delineation of the responsibilities to implement the different steps as well as a clear and stable system for the financing of these activities.

The importance of an early development of the principles and responsibilities for spent fuel and radioactive waste management has long been recognized. The IAEA has developed several safety standards and technical publications that are applicable. Guidance for development of national nuclear power programmes, including spent fuel and radioactive waste management, can be found in the so-called milestones document IAEA (2007b). More specific guidance for policy and strategy development for spent fuel and radioactive waste management can be found in IAEA (2009b). Some general conclusions are:

- A spent nuclear fuel and radioactive waste management infrastructure is a necessary element to be available when implementing nuclear power programmes.
- The development of the infrastructure requires a systematic stepwise approach lasting for several decades.
- Thus the building of the waste management infrastructure and the formulation of national spent fuel and radioactive waste policies and relevant strategies should be initiated in the early stages of planning nuclear power programmes.

14.2.2 International framework for safe spent fuel and radioactive waste management

Over the years an international regime has developed for the safe management of spent fuel and radioactive waste. Three components can be distinguished: (1) the Joint Convention on the Safety of Spent Fuel Management and the Safety of Radioactive Waste Management, (2) the IAEA Safety Standards Series and (3) the National Regulatory Control Systems.

The objectives of the Joint Convention (IAEA, 2006a) are to achieve and maintain a high degree of safety worldwide, to ensure that there are effective defences in place against potential hazards and to prevent accidents and mitigate their consequences. The Joint Convention is the first international legally binding agreement in the area of radioactive waste management. The technical basis for the Convention is provided by the IAEA Safety Fundamentals (IAEA, 2006b). It is an 'incentive' convention, which means that there are no fixed penalties and that improvements in safety are stimulated through the review process. The articles of the Joint Convention set targets. Issues covered by the Joint Convention include provisions on how to ensure safety through proper legal and regulatory systems and proper siting, design, operation and decommissioning of the necessary facilities.

The Joint Convention applies to spent fuel and radioactive waste resulting from civilian nuclear reactors and applications or handled in a civilian programme. It also includes spent sealed sources, planned and controlled releases into the environment from regulated nuclear facilities and waste from mining and processing of uranium.

The important tools of the Joint Convention are given by the review meetings that are held every three years. At the review meetings the national reports are reviewed and commented on by the parties to the Joint

Convention. The national reports give a good overview of the management of spent fuel and radioactive waste in the country. The review process provides a good opportunity for exchange of lessons learned and also encourages the countries to develop their activities. At the end of 2009 the Joint Convention had 52 contracting parties, including 26 of the 30 countries with nuclear power plants.

14.2.3 Policies and strategies for spent fuel management

Spent nuclear fuel is removed from the reactor when it can no longer contribute to the fission energy process, typically after three to seven years use. The fuel, however, still contains components, uranium and plutonium, that could be reused and recycled as fuel material. As for most waste in our society, e.g. paper and glass, there is, however, an economic issue involved in the decision to recycle or not. Although the remaining uranium and plutonium can be recycled as mixed oxide fuel (MOX) in present-day light water or heavy water reactors, real benefit from recycling will only be achieved if the fuel is recycled in fast spectrum reactors, so called fourth-generation reactors, which are being developed now. There are thus two options for spent fuel management:

- regard the fuel as a waste and dispose of it in a deep geological repository after a period (>30 years) of interim storage for sufficient cooling, or
- reprocess the fuel to separate out the components that can be recycled as fuel material after a period (~10 years or less) of interim storage. The remaining waste products (HLW and ILW) will still need geological disposal.

Some countries, e.g. Canada, Finland, Germany and Sweden,[3] have chosen the direct disposal route, while other countries, e.g. France, India and Japan, have chosen the recycling route. Most countries, however, have still not decided which option to choose. As spent fuel storage for decades is a straightforward and proven technology, there is no urgent technical need to make the choice. Prolonged storage will provide time to consider the progress in fast spectrum reactors with effective recycling, and provide a better basis for making the choice. Storage times of 100 years and more are now considered in some countries. As both options will in the end require a deep geological disposal facility, it will be important to work towards the development of such a facility, not least from a political acceptability point of view.

[3] During the 1980s these countries sent some fuel for reprocessing in Russia and France.

The views on reprocessing or direct disposal have changed over time. Some countries, e.g. Germany and Sweden that in the 1980s sent fuel for reprocessing, changed their policy in the 1980s to storage and subsequent disposal. Also in the USA the position has changed over the years. Reprocessing was the main option early on and some civilian reprocessing plants were built. Since the early 1980s the main option has been direct disposal, and investigations for developing a disposal facility at Yucca Mountain in Nevada were conducted up to the point that a licence application was submitted to the US Nuclear Regulatory Commission in 2008. This application was later recalled in 2010. In parallel, studies were conducted on reprocessing and recycling in fast reactors. In 2009 a Blue Ribbon Committee was set up to advise the Administration on the way forward. The result of the Commission is due in 2012.

The steps for spent fuel management include interim storage, reprocessing and subsequent recycling of fuel material and conditioning of the remaining waste for disposal, or encapsulation of the fuel for disposal, and final disposal. As the facilities involved are normally located at different locations, transport will also be needed. Interim storage can be made in pools in the reactor facility or in separate storage facilities, containing either water pools or dry casks or vaults (Fig. 14.1). Given the trend towards longer storage times, there is also a trend towards using dry storage systems that can be built in modules as the needs arise and that will require less active operation. There is also a trend towards primarily expanding the storage capacity at the reactor sites to avoid extra transport.

14.1 Storage of spent nuclear fuel in the Central Interim Storage Facility, CLAB, at Oskarshamn, Sweden (© SKB, photographer Curt-Robert Lindqvist).

Reprocessing facilities and facilities for producing MOX fuel exist today in only a few countries – France, India, Japan, Russia and the United Kingdom. These facilities need to be quite large and involve technology that is sensitive from a nuclear proliferation point of view. It can thus not be expected that they will be built in many countries. The existing facilities have served nuclear utilities in several more countries. In most cases the wastes from reprocessing, HLW, ILW and LLW, have been returned to the country of origin for storage and disposal.

So far no country has started disposal of spent fuel or high-level or intermediate-level waste in deep geological repositories. Development work is underway in several countries and good progress can be seen in Finland, France and Sweden, countries that expect to start disposal in the period 2020–25. Although the technology for disposal is fairly straightforward and simple, the safety assessment poses important challenges, as the time periods to be considered are very long (from thousands to hundreds of thousands of years). Another important challenge is the public and political acceptance of disposal. Important setbacks have been experienced in many countries, which has delayed the disposal projects and led to important changes in the siting process. Experience has shown that the time needed for developing a deep geological disposal facility, including the time needed for scientific studies and siting, is at least 40 years.

More technical details of the different steps for spent fuel management are given in Section 14.4.

14.2.4 Policies and strategies for management of low- and intermediate-level waste

Contrary to spent fuel and high-level waste, there is no technical advantage to delaying disposal of low- and intermediate-level waste. There is no heat production that needs to be considered, nor will the volume of waste to be disposed diminish with time. Most countries with nuclear power plants have thus developed disposal facilities. This provides the possibility of optimizing the management scheme for these wastes.

A basic principle for the management of low- and intermediate-level waste is to minimize the volumes that need to be disposed of. The first step in minimization is to avoid producing the waste, e.g. by avoiding bringing extra material like packaging into areas that are considered contaminated. Also decontamination and recycling of metals serve this purpose.

For the unavoidable waste, the management system should be designed such that it optimizes the use of resources for the whole management chain. This means that treatment and conditioning methods should be chosen to produce packages that can be handled in the transport and storage system and disposed of in the existing disposal facility. A key demand is that it should be possible to handle the waste packages as solid entities that are clean on the outside.

The management system for low- and intermediate-level waste includes sorting, treatment, conditioning and packaging systems, storage facilities and disposal facilities, and the necessary transport equipment to transport the waste between the different steps in the process. Sorting, treatment, conditioning and packaging systems are normally included at the nuclear power plants, but there are also examples of centralized or transportable conditioning facilities. For solid wastes, compaction or incineration is used to reduce the volume. Wet wastes, e.g. liquids or ion exchange resins, are solidified in packages, e.g. with cement or bitumen. In some cases ion exchange resins are stored and disposed of unconditioned in high-integrity containers.

Disposal of low-level and very low-level waste is an industrial practice, although not yet implemented in all countries, often for lack of public acceptance. Very low-level waste is disposed of in fairly simple landfills, while low-level waste is disposed of either in engineered facilities on the surface or in underground caverns. Examples of engineered facilities can be found in China, France and Spain, while underground caverns are in use in the Czech Republic, Finland and Sweden. Intermediate-level waste will be disposed of in rock caverns at a certain depth. Some facilities are under construction, e.g. in Canada and Germany.

More technical details about management of low- and intermediate-level waste can be found in Section 14.5.

14.2.5 Management of waste from the nuclear fuel cycle and from non-power applications

It is important that all types of radioactive waste in a country are considered when the policies and strategies are developed. Most countries operating a nuclear power plant or considering introducing nuclear power are likely to also have radioactive waste from non-power applications of nuclear technology, e.g. from the use of radioisotopes in medicine and research or from operating research reactors. Although the volumes of waste from such applications normally would be smaller than from nuclear power production, they often have special characteristics that need to be considered.

As has been noted above, the present practice for reprocessing companies is to return the waste separated during the reprocessing to the country of origin, which thus needs to be considered in the planning if the reprocessing route is followed. Also the other steps in the fuel cycle will generate some radioactive waste that needs to be taken care of, e.g. mill tailings from uranium mining, depleted uranium from uranium enrichment and ILW from MOX fabrication. These wastes are normally kept by the supplier. In the case of depleted uranium they are seen as a resource for future use in fast neutron reactors. If a country develops its own fuel cycle capacity, the waste also needs to be considered in the national strategy.

14.3 Radioactive waste from nuclear power production

14.3.1 Overview

During nuclear power production radioactive substances are generated as fission products, activation products and transuranic elements (which are also strictly speaking activation products). Most of the radioactivity, 99%, will be found in the spent fuel and in the structural components in the reactor core. The remaining 1% will be found in the process and technological waste, which is normally low-level waste.

The waste from nuclear power production can thus be classified as follows:

- Spent fuel elements, consisting of the fuel material (uranium oxide, plutonium oxide, fission products and transuranic elements), the fuel cladding and the structural components in the fuel element. As has been noted above, the spent fuel can also be considered a resource, as it contains components that can be further used as nuclear fuel.
- Core components, i.e. components that hold the core together and that direct the flow of water (or gas) through the core. Examples are the core grid and core barrel. Also control rods are included among the core components.
- Process waste, i.e. waste from systems used during reactor operation to clean the process water or gas or to limit the releases of radioactive substances during operation.
- Technological and maintenance waste, consisting of secondary waste generated during maintenance work and components from the reactor systems that have been replaced due to failure or wear or to renewal of the particular system.
- Decommissioning waste, with similar content to the technological and maintenance waste. It also includes the reactor pressure vessel and its internal components, which are similar to the core components.

Annually about 20–25 tonnes of spent nuclear fuel,[4] counted as uranium or uranium and plutonium (heavy metal (HM)), or 10–15 m^3, is removed

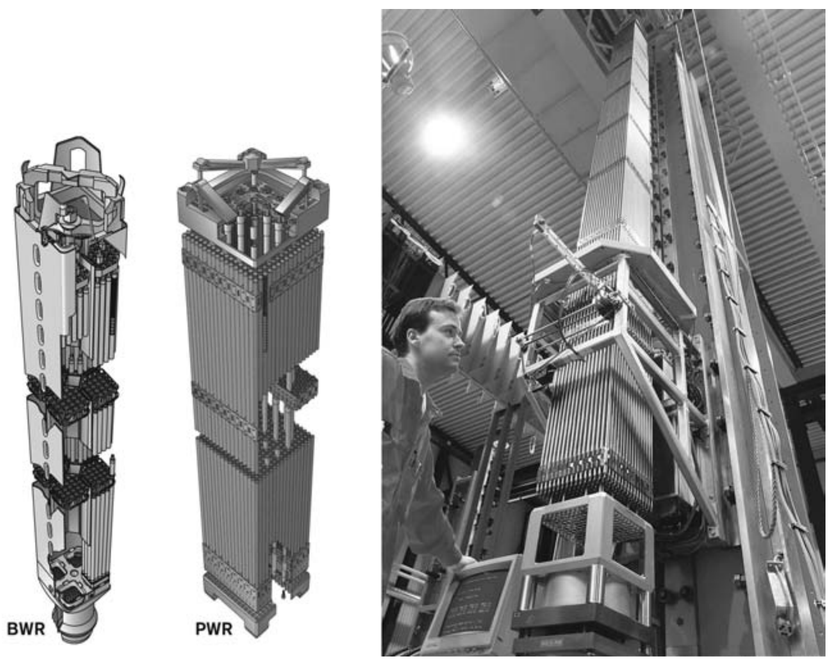
[4] The figures given in this chapter are valid for a typical 1000 MWe light water reactor. For a similar CANDU heavy water reactor which uses natural uranium fuel and thus lower burnup about 125 tonnes of spent fuel is generated with a volume of 25 m^3. With the lower burnup the activity concentration and heat release per tonne of fuel is lower. The principles for management are, however, similar.

from a 1000 MW_e light water reactor, and about 100–200 m^3 (after conditioning) of LLW is generated. The volume of ILW, mainly core components, varies depending on actions undertaken and is on average at least an order of magnitude less than the LLW.

During decommissioning a few thousand cubic metres of radioactive waste is generated. Most of this waste is VLLW and LLW, while some of the internal components are ILW.

14.3.2 Spent nuclear fuel

The fuel for current water reactors is in the form of pellets of uranium dioxide or a mixture of uranium and plutonium dioxide (MOX fuel). The uranium enrichment (content of uranium-235) is typically 3–5% in light water reactors. The pellets are very stable ceramic cylinders about 1 cm in diameter and 1 cm high. The pellets are placed in sealed thin metal tubes (e.g. of stainless steel or zirconium alloy), which are kept together as bundles to form a fuel element. The fuel element, which typically contains between 60 and 300 fuel pins, can be handled as an entity (Fig. 14.2). Fresh nuclear fuel elements need to be handled with care to avoid contamination and mechanical failures, but do not require radiation shielding. After the fuel has been used in the reactor it can still be removed and handled as an intact



14.2 Typical light water reactor fuel elements (© SKB).

fuel element. It is, however, highly radioactive due to the formation of fission products and transuranic elements in the fuel and activation products in the fuel element structure. The typical composition of spent fuel (excluding the fuel element structure) is:

- 95% uranium (remaining enrichment about 0.8%)
- 1% plutonium
- 4% fission products and transuranic elements other than plutonium.

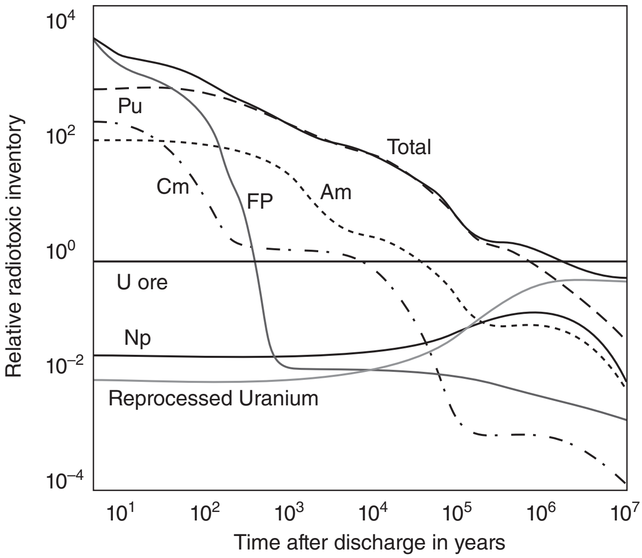
A more detailed composition of a typical LWR fuel element is given in Table 14.1. Some of the fission products are very short-lived with half-lives of a year or less, while others have half-lives ranging from 30 years to millions of years.

The spent fuel element has a high concentration of different radionuclides that decay by emitting α-, β- or γ-radiation or undergo spontaneous fission that emits neutrons. The α- and β-radiation is mainly absorbed in the fuel itself and is the energy dissipated as heat (decay heat), while the γ- and neutron radiation is more penetrating so that the spent fuel will require shielding. Some neutrons also generate additional fission in the fuel, which will require control of the spent fuel configuration to avoid criticality. The spent fuel thus needs shielding and cooling during the subsequent handling. After removal from the reactor the fuel is stored under water for several years to allow cooling. During the first year the decay heat goes down rapidly as the short-lived fission products decay. After about five years the decay heat is dominated by cesium-137 and strontium-90, which both have a half-life of about 30 years.

Spent fuel remains radioactive for very long times, hundreds of thousands of years, and will eventually need final geological disposal to ensure long-term isolation from humans and the environment. In Fig. 14.3 the radioactive decay for spent fuel is shown. The curve shows the toxicity index, which takes into account not only the activity but also the harm the radioactive substance would give if incorporated into the body (essentially eaten). After the first few years the toxicity is dominated by cesium and strontium. After a few hundred years the toxicity will be dominated by the transuranic elements, such as plutonium and americium. By removing plutonium and possibly also some other transuranic elements the long-term toxicity and heat release can be reduced, but it is generally considered that long-term geological isolation will still be needed.

14.3.3 Waste from reprocessing

The main reason for reprocessing is to separate the remaining uranium and plutonium in the fuel from fission products and transuranic elements other than plutonium, so that these materials can be reused as material for new



14.3 Relative radiotoxicity of the different components in spent nuclear fuel from a light water reactor irradiated to 41 MWd/kg U with respect to the radiotoxicity of the corresponding uranium ore (NEA, 1999c).

fuel (plutonium mixed with uranium in Mixed Oxide (MOX) fuel and reprocessed uranium re-enriched in so called 'reprocessed uranium' (REPU) fuel). In this process also the volume of high-level waste and the long-term toxicity is reduced. By more advanced reprocessing, that is not yet in use, it could also be possible to remove the transuranic elements and some long-lived fission products from the waste, thus further reducing the radiotoxicity. The intention would then be to burn (nuclear incineration) the removed components in a fast neutron reactor or another fast flux nuclear facility, e.g. an accelerator driven reactor (ADS).[5]

The waste from the reprocessing includes the high-level waste containing the fission products and transuranic elements other than plutonium, the metal components of the fuel element (fuel cladding, end pieces and spacers), and secondary process and maintenance waste. In the end decommissioning waste will also be generated.

The solution of fission products and transuranic elements is concentrated and then mixed with glass-forming components and melted to become a

[5] In an ADS a subcritical reactor configuration is connected to a particle accelerator. The accelerated particles impinge on a heavy target (e.g. lead) and generate a burst of neutrons that will produce fissions in the subcritical reactor, thus generating energy. The neutrons can also fission some long-lived transuranic elements, thus producing more short-lived radionuclides.

glass matrix (vitrification), which is poured into a metal container that is subsequently sealed and kept clean on the outside. This is the main HLW. Most of the radioactivity of the fuel remains in the HLW.

The metal components of the fuel element are further cleaned to minimize the remaining fuel oxide in this waste stream. After compaction or cementation the metal components are filled in tight containers similar to the HLW and handled in a similar way as HLW or ILW.

The process and maintenance waste is physically similar to such waste from a nuclear power plant (see Section 14.2.4). There is, however, an important distinction as the fuel is dissolved in the reprocessing plant and the systems and components are exposed directly to the fuel material. Some of this waste could thus contain significant amounts of long-lived radionuclides and would therefore be classified as ILW.

Reprocessing of one year's fuel from a 1000 MW_e LWR will generate 2–3 m^3 of vitrified HLW and some 10 m^3 of ILW and LLW. Some ILW and LLW will also be generated in the MOX fabrication facility.

14.3.4 Core components

Core components normally have a longer operational life than the nuclear fuel and are only removed when the structural integrity is reduced by cracking, corrosion or ageing phenomena or, in the case of control rods, when their neutron absorption capacity has been reduced. The radioactivity in the core components is mainly activation products, the most important from a handling point of view being cobalt-60. From a disposal point of view there are also some long-lived nickel and niobium isotopes and the core components are generally considered to be ILW.

The radiation level from the most exposed core components is similar to or higher than from the spent nuclear fuel, but the heat generation is lower and it decays more rapidly.[6] For practical reasons the core components are initially handled in a similar way to the spent nuclear fuel and stored in the reactor spent fuel pool.

14.3.5 Process waste, technological and maintenance waste

Radioactive waste is continuously generated during the operation of a nuclear power plant. Process waste comes from the continuous clean-up of the coolant that is circulated through the reactor core. It also comes from

[6] The radiation level from core components comes mainly from γ-emitting radionuclides that are not absorbed in the core components material itself. Therefore the heat generation is lower.

the control of releases of water and gas from the reactor facility. Clean-up is achieved by mechanically filtering the water and by ion exchange to demineralize the water. The purpose of the clean-up of the process water is to create a chemically benign environment to reduce corrosion and build-up of debris on the fuel (crud) and also to reduce the source for activation products that can spread through the reactor systems. The purpose of control of releases is to ensure that only small amounts of radioactivity, which are well within the regulatory limits, are released from the reactor facility. The primary waste products produced in the process waste are filter cartridges and sludge of ion exchange resins, other filter material and evaporator concentrates. The filter cartridges can be handled in a similar way to other technological waste while the sludge needs solidification before further handling as waste. The activity concentration depends on the processes used and the location of the filtering systems in the reactor. Particularly high activity would be found in the ion exchange resins in the primary system. The main radioisotopes are corrosion products, e.g. cobalt-60 and iron-55, and fission products, e.g. cesium-134, cesium-137 and strontium-90. The amount of fission products depends on the integrity of the fuel.

Technological and maintenance waste consists of exchanged components and material (e.g. paper, coveralls, discarded instruments, scaffolding and oils) that is used during maintenance. In most cases the activity concentration is very low in technological and maintenance waste. It contains the same radioisotopes as the process waste. Some exchanged components can have higher activity concentration. This can be reduced by mechanical or chemical decontamination as it is mainly surface contamination.

14.3.6 Decommissioning waste

A closed reactor can in its entirety be seen as waste, and the decommissioning and dismantling process can be seen as waste management. A key component in the management process is to segregate non-radioactive waste from radioactive waste. The larger part of the reactor, e.g. buildings and systems that have not been in contact with process waters or gases, can be regarded as non-radioactive waste and be taken care of like normal industrial waste. The remaining radioactive waste covers a wide spectrum of types and activity concentrations, ranging from core components with a very high radiation level to very low-level waste, similar to the maintenance waste.

A key component for successful decommissioning and dismantling is an effective and well-planned waste management system including choosing the right size of the waste packages and the right level of decontamination. The optimal levels will differ between countries depending on their entire waste management system, in particular the transport system and disposal facility and the possibilities of recycling decontaminated material. Ideally, recycled material should be used without restrictions, but also some material with a radioactivity concentration above the release limit could be recycled in the nuclear industry for waste packages or some reactor components.

14.4 Management systems for spent nuclear fuel

14.4.1 Overview

As described earlier in Section 14.2.3, there are different ways of looking at spent nuclear fuel, either as a resource to be reprocessed and recycled or as a waste to be disposed of in a geological repository. In the case of reprocessing the valuable materials, plutonium and uranium will be recycled as MOX or REPU fuel and the remaining waste will need geological disposal.

The choice between the two options, recycling or disposal, will be based on strategic, political and economic factors. At present about 15–20% of all spent fuel is reprocessed and the plutonium and uranium recycled. The recycling takes place in light water reactors. Although over the years recycling has been performed in several countries, it is primarily France that is doing it on an industrial level today. France has both reprocessing and MOX fabrication capacity. Other countries such as Japan, Russia and China are preparing for recycling. Recycling will lead to a better utilization of the natural uranium resource. Recycling in light water reactors will reduce the uranium consumption by about 25%. For more effective use, recycling in fast reactors will be necessary, which will allow multiple recycling of the generated plutonium and which can also utilize the depleted uranium from enrichment, which otherwise would be a waste product, as breeding material for new plutonium that can be utilized as fuel. Theoretically, recycling in fast reactors could lead to the uranium being utilized at least 50 times more effectively, i.e. one could get 50 times more energy out of the natural uranium. Fast reactors are, however, not yet available on a commercial scale for electricity production. Important development work is going on in several countries, e.g. Russia, France, India, Japan, China and the United States. Except for India, it is, however, not expected that recycling in fast reactors will be of significant importance before 2050. The technical and economic feasibility still needs to be proven before fast reactors can be introduced on a commercial scale. From a waste management point of view there could also be an additional added value in recycling in fast reactors as this could provide the possibility to also burn (transmute into shorter-lived elements that can be disposed of more easily) some of the other transuranic elements, e.g. americium and curium, elements which make an important contribution to the long-term radiotoxicity of the fuel (see Section 14.3.2).

Although the management system for spent nuclear fuel will be different depending on what management strategy is chosen, recycling or disposal, there are also many components in common.

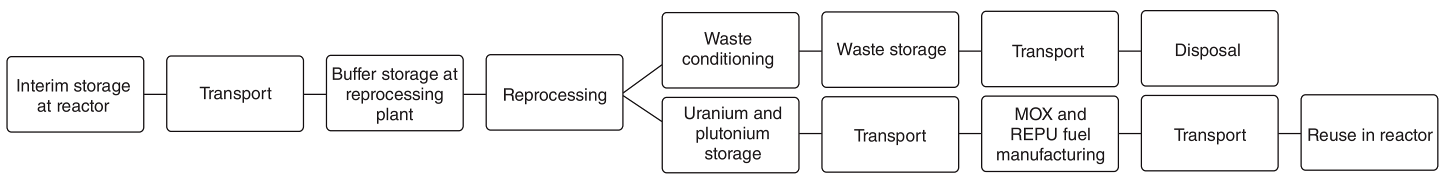
The management system for recycling of the spent nuclear fuel includes the steps shown in Fig. 14.4. Reprocessing is normally performed 3–10 years after the fuel has been removed from the reactor. In principle the fuel can be transported away from the reactor already after about a year, but often the transport is later. At present it is not foreseen to recycle the MOX fuel again in light water reactors, but to store it for later use in fast reactors. It should, however, be realized that the spent MOX fuel has a higher heat generation, radiotoxicity and neutron radiation level than the corresponding spent uranium fuel.[7] This will be important should the spent fuel later be considered for direct disposal and not for recycling.

The management system for direct disposal of spent fuel has the components shown in Fig 14.5. The cooling time before spent fuel can be disposed of in a geological repository will be typically 30–50 years or longer.

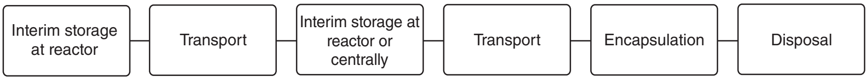
Over the years approximately 400,000 tonnes of spent fuel (measured as heavy metal (HM)) have been generated. About 100,000 tonnes of these have been reprocessed and the remaining 300,000 tonnes remain in the reactor pools or are stored in dedicated facilities within the power plant premises or in centrally located storage facilities either for direct disposal or awaiting a later decision to reprocess. More details about different storage facilities at the reactor site or centrally are given in Section 14.4.2.

Except for the reprocessing step, spent fuel management has so far been mainly a national activity. Storage facilities are built at the reactors or centralized in the country. No international storage facilities have been developed. The same is the case for the work on geological disposal. Although there is much international cooperation on research and development for geological disposal, there are no agreements between countries to develop a common geological disposal facility, in spite of the technical/economic advantages it could bear. Discussions have taken place in different fora but no real progress has been seen, mainly due to the political sensitivity of the subject. Reprocessing is the exception. Several countries have jointly financed some reprocessing plants and sent their fuel for reprocessing to these plants. The agreements have, however, included the stipulation that the waste from reprocessing will be returned to the country of origin for further management and disposal.

[7] The MOX fuel has a higher concentration of transuranic elements from absorption in different plutonium isotopes.



14.4 Management system for recycling of the spent nuclear fuel.



14.5 Components of the management system for direct disposal of spent fuel.

14.4.2 Spent fuel storage

Irrespective of which option is chosen, the first step in the spent fuel management is interim storage. The spent nuclear fuel element is mechanically the same as fresh fuel and can be handled as an intact fuel element. It is, however, highly radioactive and needs shielding and cooling during handling. The shielding is mainly for gamma and neutron radiation from the fuel. The spent fuel element is thus, after removal from the reactor vessel, handled and stored under water that can provide adequate shielding and cooling.

All water-cooled reactors store the spent nuclear fuel in deep water-filled pools. The water depth is typically 10 metres or more to ensure that the water provides adequate shielding (3–4 metres coverage) during all handling of the fuel. As the original intention in many cases was to reprocess the fuel, the size of the pools was designed to store a few years' production only in these pools. In later reactors larger storage capacity has been provided, in some cases corresponding to 30 years' production or more.

As reprocessing currently is used in only a few programmes, it has been necessary to expand the storage capacity for most reactors. Different methods have been used. In some cases it has been possible to pack the fuel closer in the existing pools by introducing neutron absorbers or by taking into account the fact that the reactivity of spent fuel is lower than that of fresh fuel for which the storage racks in the pools were designed (burn-up credit). In other cases new storage facilities have been built, either at the reactor site or at a central site away from the reactor, to which the fuel can be transferred. In some cases these are also built as deep storage pools, while in other cases the fuel is stored under dry conditions in metal or concrete casks similar to transport casks, or in vaults or silos. Dry storage can be used when the heat release has diminished sufficiently after 5–10 years of storage. Although there are several wet-type storage facilities in operation, the trend at present is to use dry storage facilities for long-term storage. These have the advantage of requiring less long-term maintenance and can also more easily be expanded as the needs arise. The most obvious example of the latter is the dry storage casks, which essentially can be purchased as they are needed (Fig. 14.6). A typical modern storage cask can accommodate 20–40 PWR fuel elements or 50 – 100 BWR fuel elements, which means that a large reactor will require only about two to three casks

14.6 Storage of spent nuclear fuel in dry storage casks (Gorleben, Germany).

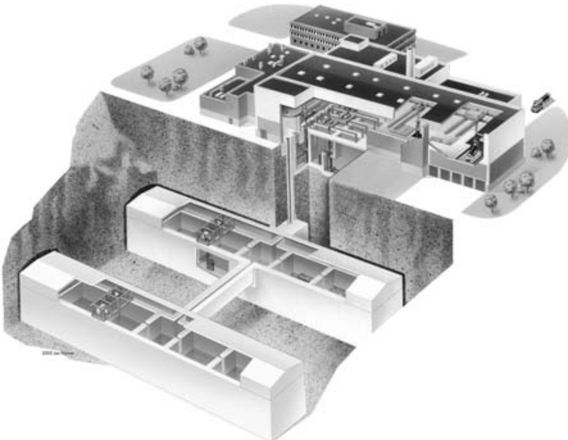
per year. The casks can be stored in simple warehouses and do not require strong buildings. There are also proposals for multipurpose casks that can be used for storage and transport and possibly even disposal. An overview of existing storage types is given in IAEA (2007c).

There is ample experience of long-term storage, up to 50 years, of spent fuel in water. So far no degradation of the fuel has been seen if the water quality is kept under control. The experience of dry storage is also good, although for a shorter period (less than 30 years). It is expected that the storage times can be extended without problems to at least 100 years, but the proof of such extension will require some further studies, in particular to ensure that the fuel can be removed after such a long period.

The main difference between wet storage and dry storage is the need for continuous cooling and chemical clean-up of the pool water to ensure a low fuel temperature and avoid long-term corrosion of the fuel or the spent fuel pools. Wet storage thus normally requires more staff for operation. Dry storage represents a much higher fuel and fuel cladding temperature, but a more benign environment, normally helium or argon gas. It is, however, important to have a follow-up programme of the fuel to ensure that the fuel can still be removed from the dry storage when the fuel is transferred to the next step. Most storage facilities are built above ground, but like the Swedish CLAB facility they could be built in a rock cavern to provide a better physical protection over long time periods (Fig. 14.7).

14.4.3 Spent fuel transport

The transport of spent nuclear fuel and radioactive waste is regulated by national authorities and based on the IAEA transport recommendations



14.7 Spent fuel storage pools in rock chambers at the Swedish Central Interim Storage Facility, CLAB, at Oskarshamn, Sweden (© SKB).

(IAEA, 2009c). For spent nuclear fuel so-called type B packages will be needed, that can sustain drops, fires and submersion in water (more details of these tests are given in the transport recommendations). A transport cask for spent fuel is typically a cylinder about 5 m long and 1–2 m in diameter. It is designed to provide adequate shielding against gamma and neutron radiation, to control criticality and to remain tight in case of postulated accidents. The weight is around 50–100 tonnes (Fig. 14.8). There is ample experience of spent fuel shipments from nuclear power plants to reprocessing plants or to central storage facilities (more than 100,000 tonnes). Most transports of spent fuel have been made by rail or ship, but also shorter transports on normal roads have been made. Transports have been made within countries such as France, Russia, the United Kingdom and the USA and also across borders, e.g. from Finland and Bulgaria to Russia, and from Japan and Germany to France and the United Kingdom.

14.4.4 Spent fuel reprocessing and recycling

The main reason for reprocessing is to separate the remaining uranium and plutonium in the fuel from fission products and transuranic elements other than plutonium, so that these materials can be reused as material for new fuel (MOX fuel with plutonium mixed with uranium or REPU fuel with reprocessed uranium).

During reprocessing the spent fuel is dissolved in hot nitric acid and the solution is subsequently exposed to several chemical processing steps to

14.8 Transport container TN17 for spent nuclear fuel (© SKB).

separate the different components. In the present reprocessing facilities four main product streams can be distinguished:

- Uranium
- Plutonium
- Fission products and transuranic elements other than plutonium
- The metal components of the fuel element (fuel cladding, end pieces and spacers).

The uranium and plutonium are purified such that they can be either reused as MOX fuel or re-enriched to form REPU fuel, while the waste streams are treated and conditioned as described in Section 14.3.3.

At present two large reprocessing facilities, La Hague in France and Sellafield in the UK, are in operation, with a capacity of 1600 and 800 tonnes of spent fuel per year (measured as heavy metal (HM)) respectively. A third large facility (800 tonnes HM/year) is in pre-commercial testing at Rokkasho in Japan. Smaller reprocessing plants (100–400 tHM/year) are in operation in Russia, India, Japan and China. Approximately 15–20% of the spent fuel being generated today is reprocessed. The remainder is stored for direct disposal or a future decision to reprocess.

Reprocessing is a proven industrial technology. Development work is going on to increase the proliferation resistance (e.g. by not producing separated plutonium). Recycling of the plutonium as MOX fuel in light water reactors as well as the reprocessed uranium is also performed on a routine basis, in particular in France. The economy of reprocessing and recycling in LWR will differ from country to country. The situation is quite different for a country with its own reprocessing facility than for a customer country. Some countries also have political concerns about reprocessing. All in all this has led to the situation that today reprocessing plants are not fully utilized and most countries have adopted a wait-and-see position.

The increasing expectations for nuclear power use in the future have, however, revived the interest in reprocessing and recycling. Several initiatives have been launched over the last few years to increase the international cooperation in this field, e.g. the International Project on Innovative Nuclear Reactors and Fuel Cycles (INPRO) (IAEA, 2010), the Generation IV International Forum (GIF, http://www.gen-4.org/), and the International Framework for Nuclear Energy Cooperation (IFNEC, earlier the Global Nuclear Energy Partnership (GNEP)).

For nuclear energy to be sustainable in the long term (more than a few hundred years) it will be necessary to introduce at some time fast reactors that will utilize the uranium resource in a more efficient way. The real economic value of recycling will only come with the development of fast reactors. The important question for spent fuel management is when fast reactors will be introduced such that recycling can have a real impact.

The waste from reprocessing, i.e. HLW containing fission products and transuranic elements, and ILW containing the metal components of the fuel elements and secondary waste, will require geological disposal after conditioning. The heat generation from the HLW needs to be considered in the design of the repository. Development work is going on to also separate out the transuranic elements during reprocessing to reduce the long-term heat generation (over >100 years) and also the radiotoxicity of the high-level waste (advanced reprocessing). This would have the potential to simplify the design of the repository and the long-term safety assessment (although the transuranic elements rarely are dominating the doses in the safety assessment). To achieve this gain, the separated transuranic elements will need to be recycled and burned in a fast reactor system. As for fast reactors, this development will require at least another 50 years for commercial introduction.

14.4.5 Storage and transport of waste from reprocessing

The HLW from reprocessing has a high radiation level and generates heat. It will thus require special care for storage and handling. The HLW canisters are stored at the reprocessing plants in large vaults. Normally the HLW canisters are stacked on top of each other (about 10) in tubes that are cooled by air from the outside. Transport of the HLW is carried out with transport containers similar to those used for spent fuel. These containers can also be used for storage of the waste.

The ILW and LLW from reprocessing has a very low heat generation and can be stored and transported in a similar way to ILW and LLW from nuclear power plants (see Section 14.5.3).

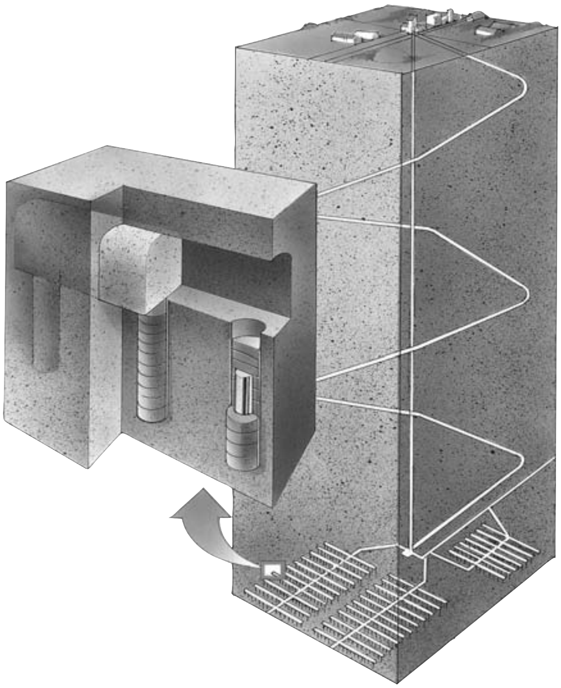
14.4.6 Disposal of spent nuclear fuel and waste from reprocessing

Spent fuel and high-level waste from reprocessing is highly radioactive and remains dangerous for thousands to hundreds of thousands of years and will require isolation in a deep geological repository. Also the ILW from reprocessing will require disposal in a geological repository. So far no such repository has been built. There is, however, an international consensus among the experts that disposal in a deep geological repository can be made in a safe way and that the long-term safety can be assessed (NEA, 1999a, 1999b, 2009; Witherspoon and Bodvarsson, 2006). Several countries are developing the design for a deep geological repository and are in the process of looking for a suitable site. Good progress is being made in Finland, France and Sweden. Finland and Sweden have decided to dispose of the spent fuel directly. Sites have been chosen and the licence is under preparation. Operation is foreseen to start shortly after 2020. France will dispose of high-level vitrified waste and other long-lived reprocessing waste at depth in a clay formation in eastern France. The detailed siting is going on and operation is planned for around 2025. The progress of all these three projects will be very important as it will show the feasibility of disposal irrespective of whether the fuel is reprocessed or not.

The principles employed for geological disposal are fairly simple. The waste, which in itself is a solid that is resistant to dissolution and leaching, is placed in a tight container that is designed to remain tight for a long time and the container is placed in an environment that is benign for keeping the tightness. To ensure the latter, the siting looks for a geological medium that can be expected to remain mechanically and chemically stable for the long time periods required. In particular the chemical processes at depth, with small water movements, are very slow.

The safety of waste disposal is based on the multibarrier principle, i.e. the waste shall be surrounded by several barriers that are functioning independently of each other. This means that if one barrier fails, or our knowledge of the processes affecting the integrity of that barrier is not correct, the other barriers will ensure the long-term safety.

In Fig. 14.9 the disposal concept (KBS-3) developed in Sweden and Finland is shown. The spent fuel is encapsulated in copper canisters that are stabilized by an internal iron structure. The waste canisters are placed in boreholes at the bottom of tunnels at about 500 m depth in the granitic rock found in Sweden and Finland. In the boreholes the canisters are surrounded by bentonite clay, which provides a mechanical and chemical protective buffer. At the end also the tunnels are backfilled with a mixture of bentonite and sand. This is the system that will be used for the first geologi-



14.9 Schematic presentation of the KBS-3 disposal system for spent nuclear fuel to be implemented in Finland and Sweden (© SKB, illustration by Jan M. Rojmar – Grafiska Illustrationer).

cal disposal facilities that are considered in these countries for disposal of spent nuclear fuel. The barriers are:

- The fuel matrix itself, in which most of the radioactive elements are part of the matrix and are released only when the matrix is dissolved or corroded. The dissolution/corrosion rate is very low in the kind of water existing at the repository depth.
- The copper canister, which is highly corrosion resistant in the chemical environment created by the bentonite and the reducing groundwater at the repository depth. The iron structure in the canister ensures the mechanical stability of the canister against the pressures found at depth from the rock, the groundwater and the swelling bentonite clay.
- The bentonite clay, which reduces the inflow of corrodants from the ground water to the canister and also ensures that the outflow of radionuclides from the fuel, if the tightness of the canister is broken, is very slow. The bentonite also has a chemical buffering effect, keeping a stable pH.

- The surrounding rock, which has a low water flow and ensures that the transport of corrodants to the bentonite buffer and the canister is slow. The rock also provides mechanical stability around the canister. Finally the rock acts as a filter if the tightness of the canister is broken and radionuclides are transported out from the fuel through the canister and the bentonite. The filtering function has two components. First, the transport of water is very slow, thus providing time for radiological decay of the radionuclides, and second, the transport is further delayed by chemical adsorption of the radionuclides at the surfaces of the cracks through which water and radionuclides are transported.

Similar disposal systems are being being considered in other countries but have to be adapted to the specific geological settings chosen and to the waste forms. Different geological media are being considered in different countries. In addition to hard rock like granite, also clay and salt formations as well as sedimentary rocks are being investigated. As noted above, a clay formation has been chosen in France, for example, and disposal in salt has been the main line of investigation in Germany. Also different canister materials are being considered.

The disposal of ILW could in principle be based on a simplified version of the basic principles for HLW disposal, as the main concern for this waste could be human intrusion disposal at less depth, e.g. 100 m is considered for ILW. A repository for ILW has been in operation in the USA since the mid-1990s. It is the Waste Isolation Pilot Plant (WIPP) in Carlsbad, New Mexico. Here the waste is disposed of in a dry salt formation in large salt rock chambers that are subsequently backfilled with crushed salt (Fig. 14.10). Another ILW repository is under construction in Germany at the Konrad mine.

An important component in the development of a deep geological disposal facility is the study of different technologies and processes in underground research facilities. Several such facilities have been developed around the world, e.g. the HADES facility in clay in Belgium, the URL in granite in Canada, the Äspö laboratory in granite in Sweden, and Grimsel in granite and Mont Terri in claystone in Switzerland.

Waste disposal is not only a technical question, it is a highly political and societal question and requires a strong commitment from society as well as from the industry. In several countries there have been political setbacks delaying programmes. The most spectacular ones have been in Germany and the USA. In Germany the development towards a repository in the Gorleben salt dome was well underway in the 1990s when it was halted by a political decision on a 10-year moratorium to investigate alternatives. At the time of writing this book (September 2010) discussions are underway to resume the work in Gorleben. In the USA a decision was made several

14.10 The Waste Isolation Pilot Plant uses a continuous miner to carve disposal rooms out of the Permian Salt Formation, nearly a half mile below the surface (© US DOE).

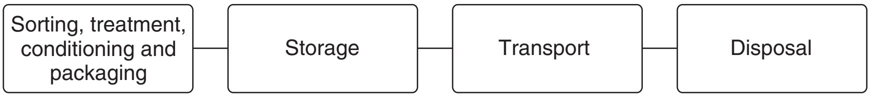
years ago to develop a repository at Yucca Mountain in Nevada. Following many years of costly investigations and the preparation of an extensive licence application, a political decision was, however, made in 2009 to bring the project to a halt, although at the time of writing the final fate of Yucca Mountain is not yet determined. These examples show that politics can easily cost much more than engineering.

14.5 Management of low- and intermediate-level waste

14.5.1 Overview

Contrary to spent nuclear fuel and high-level waste, there is no need to put LLW and ILW in interim storage for heat decay. This waste can be disposed of directly after it has been conditioned and packaged if a disposal facility is available. Some buffer storage is, however, normally built at the reactor site. The steps for management of LLW/ILW are shown in Fig. 14.11. The first step is normally performed at the power plant and results in a package that is clean on the outside and can be further handled in the subsequent steps. In some cases also, centralized treatment and conditioning facilities have been erected, e.g. for incineration or melting of low-level waste.

A central principle for the management of LLW and ILW is waste minimization in both activity and volume by appropriate design measures and operating and decommissioning practices. A key component is the selection and control of materials used in the nuclear power plant or during



14.11 Steps for management of LLW/ILW.

maintenance. Bringing unnecessary material into the radiologically controlled zone should in particular be avoided as this might later be declared as radioactive waste. It is further recommended to segregate the waste produced at the source to avoid cross-contamination of low-active material with material with higher activity. Other methods for waste minimization are decontamination of the waste for recycling to the extent possible and economically justified and compaction and/or incineration of compactable and combustible waste.

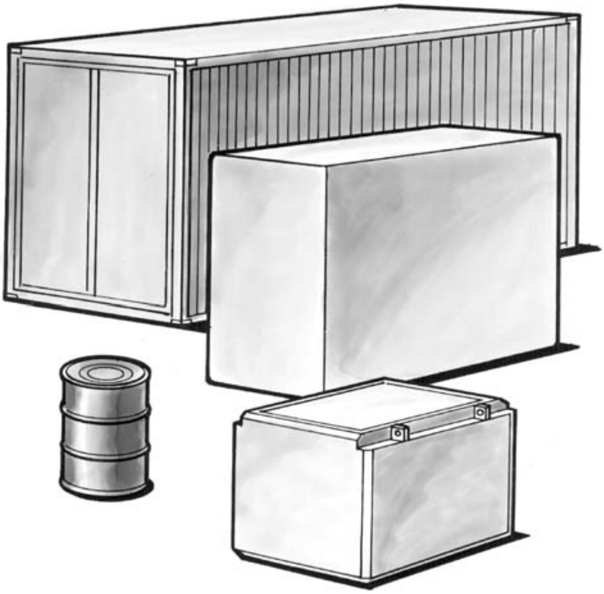
14.5.2 Treatment and conditioning of LLW and ILW

As described in Section 14.3, a wide variety of radioactive waste is generated during the operation and maintenance of a nuclear power plant and during the final decommissioning and dismantling of the reactor. They are generally distinguished as wet and solid waste. In addition also gaseous waste is generated. These wastes are released to the atmosphere after appropriate filtration and the filters containing the important radioactivity can be handled as solid waste.

The purpose of the treatment and conditioning is to produce a waste package that is suitable for the subsequent waste management steps, i.e. storage, transport and disposal (Fig.14.12). Another purpose is to reduce the volume as is much as is economically justified.

Liquid waste, i.e. contaminated water, is treated by chemical precipitation, ion exchange, mechanical filtering and/or evaporation depending on the concentration of radioactivity in the water and the cleanliness of the water, as well as the further use for the water. The products from these treatment processes are wet sludge (solid content <15%), spent ion exchange resins, and filter cartridges. The sludge and ion exchange resins are normally then conditioned to form a solid body (solidification) directly in a package suitable for handling and disposal, while the filter cartridges can be handled as solid waste.

The methods most commonly used for solidification of wet wastes are cementation, bitumination, polymerization and vitrification. In the cementation process, which is the most widely used method, the waste is mixed with cement to form a concrete that is poured directly into the waste package. Care must be taken that the chemical composition of the waste is compatible with the cementation process. The process is fairly straightfor-



14.12 Different types of waste packages for low- and intermediate-level waste.

ward and is widely used in most nuclear power plants. A drawback is that it normally leads to a volume increase. In the bitumination process wet wastes are mixed with hot bitumen and the remaining water is driven off, thus providing a volume reduction. The bitumen/waste mixture is then filled in the final waste package, normally a 220-litre drum. Bitumination is used at some nuclear power plants and also some reprocessing facilities. More recently also polymerization and vitrification technologies have been developed for wet LLW and ILW. In the polymerization process the waste is mixed with a polymer that uses the excess water for polymerization. In the vitrification process the waste is heated together with glass-forming components to create a radioactive glass. Vitrification normally requires that the wet wastes are pre-dried. The different processes considered for conditioning of wet wastes have different advantages and disadvantages. The choice of process will depend on many factors such as the volumes to be treated, the activity concentration, the chemical composition, the requirements from the disposal facility and the end costs.

Solid waste has a wide variation in physical form and activity content and the chemical form of the activity. Metal waste is normally decontaminated with mild acids such that the material can be reused. When a material is finally declared as waste it is then separated into combustible, compactable or non-compactable waste. Combustible waste can be incinerated and the ashes taken care of as wet or solid waste and the filters as solid waste. An incineration facility for radioactive material is, however, an expensive installation and will normally require a substantial volume of waste to be incinerated. Most of the incinerators installed are therefore central for a country, e.g. in France and Sweden. For compactable waste different types of compactors are used, ranging from simple drum compactors, where the waste is compacted directly in the waste package drum, to high pressure (>1000 Mg) supercompactors. In the supercompactors standard 220-litre drums with waste are compacted to form thin 'slices' that can then be packaged in a drum for subsequent handling. To stabilize the compacted waste or non-compactable waste in the final waste package, concrete is normally poured into the package to provide a solid monolith.

An important part of the waste treatment and conditioning processes is waste characterization. It has to be ensured that the waste form is suitable for the next step, e.g. has the suitable chemical form and/or activity concentration. It is particularly important that the conditioned waste package will fulfil the requirement, i.e. the waste acceptance criteria, for transport and disposal.

In most cases each nuclear power plant is equipped with the appropriate facilities for waste handling, treatment, conditioning and storage. In some countries centralized facilities, e.g. for incineration, have been built. In other countries mobile treatment and conditioning facilities have been introduced, that can serve several nuclear power plants.

14.5.3 Storage and transport of LLW and ILW

The waste packages that are produced have been adapted to the requirements for storage and transport as well as for disposal. Different types of packages are used. The most common are standard 220-litre steel drums or standard 10- or 20-foot shipping containers. Other types of containers are steel packages of other sizes and packages with a concrete wall that provides some shielding. The packages are normally clean on the outside so that the further handling can be made without the need to consider contamination. The packages, however, still emit radiation that needs to be considered during the handling. In many cases the radiation level is such that the packages can be handled, stored and transported without extra shielding, i.e. the packages fulfil the transport regulations. For waste with a higher activity concentration, the dose rates from the waste packages are higher and they will need extra shielding during handling, storage and transport.

LLW and ILW can be stored in fairly simple warehouse-type buildings. Normally the walls are made of concrete of appropriate thickness to provide shielding for the outside.

Transports of LLW and ILW need to fulfil the transport requirements. LLW packages that by themselves fulfil the requirements can be transported in simple standard shipping containers, while packages with a higher dose rate will need to be transported in sturdy thick-walled containers. In many cases it should be enough to fulfil the requirements for so-called type A containers, while in some cases with a higher activity concentration type B containers will be needed (IAEA, 2009c).

LLW and ILW can be transported in a similar way to spent fuel and high-level waste on trucks, trains or ships, depending on the locations of the nuclear power plant and the repository.

14.5.4 Disposal of LLW and ILW

LLW is defined as waste that contains only limited amounts of long-lived radionuclides, but still requires robust isolation and containment for periods up to a few hundred years. It is suitable for disposal in engineered near-surface facilities. ILW has a radioactivity content that requires disposal at greater depths, of the order of tens of metres to a few hundred metres. Disposal of ILW was discussed in Section 14.4.6.

Disposal facilities for LLW have been in operation for more than 20 years in several countries around the world. Some of the earlier disposal facilities had a very simple design and the waste was essentially disposed of in trenches above the water table and with a watertight cover. The more modern disposal facilities have a more engineered design with several barriers against release. Two different types can be distinguished, engineered surface facilities and engineered facilities in rock chambers. Both types of facilities can superficially be described as disposal in a house that should remain tight for water intrusion, but which still has control over any water coming out from the house. In all cases the multiple barrier approach is being used to ensure long-term containment of the radioactive elements.

Two examples of near-surface engineered facilities are the Centre de Stockage de l'Aube (CSA) in France and El Cabril in Spain, which have been in operation since the early 1990s. Similar facilities are in operation or under construction in several other countries, e.g. Japan, China and Belgium.

In CSA the disposal is made in large concrete structures ($25 \times 25 \times 8$ m) that are built on the surface (Fig. 14.13). The conditioned waste packages are placed in the concrete structures and subsequently surrounded by concrete. When one concrete structure is filled a reinforced concrete lid is cast, including an impermeable cover. The disposal operations take place under a temporary roof that can be moved from disposal structure to disposal structure. Underneath the concrete structure there is a channel system for collection and control of any water that might come out of the structure.

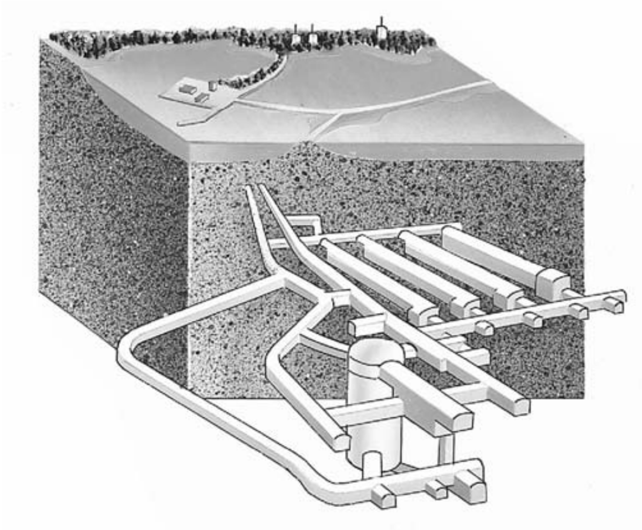
14.13 Aerial view of the Centre de l'Aube disposal facility for low-level waste in France (© 4 vents).

Each concrete structure can house about 2500 m^3 of conditioned waste. The whole CSA site is designed for 1,000,000 m^3. After completion of the disposal the concrete structures will be covered by clay and earth and grass will grow on top of the mounds thus made (Fig. 14.14). The site is intended to be surveyed, including control of any effluents, for at least 300 years, i.e. approximately 10 half-lives for cesium-137 and strontium-90. The long-term safety of the disposal (>300 years) is based on the low content of long-lived radioelements, the characteristics of the waste form and packages, the watertight concrete structure and finally the surrounding geology with a low water flow.

LLW disposal facilities in rock chambers at about 100 metres depth are in operation at Olkiluoto and Loviisa in Finland and Forsmark in Sweden (SFR). Other similar facilities are under construction in Korea. In some countries disposal of LLW is planned at greater depth, e.g. in Germany (Konrad) and Canada (Bruce).

In SFR the repository has been placed between 50 and 100 metres below ground level. It consists of several different rock chambers that have been adapted to the type and activity level of the waste (Fig. 14.15). Some very low-level waste is disposed of directly in the rock chambers with no extra barriers than the waste package itself and the rock, while the more active LLW is placed in a large concrete silo (50 m high, 50 m diameter) and surrounded by concrete. Between the concrete silo wall and the rock a buffer of bentonite clay is introduced to further reduce any leakage. The multiple barriers are thus the waste form and package, the concrete structures, the bentonite clay and the rock. The facility has been built with the intention of making it possible to abandon it without further surveillance once it has

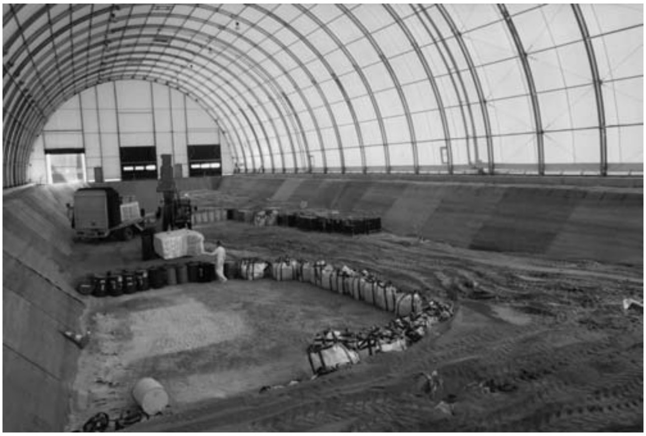
14.14 Aerial view of the Centre de la Manche disposal facility in France. The disposal facility has been closed and covered with clay and grass (© Zorilla Production).



14.15 Cut-away view of the Swedish Disposal Facility for Low Level Waste, SFR, at Forsmark, Sweden (© SKB, illustration by Jan M. Rojmar – Grafiska Illustrationer).

been filled. Whether this will happen in reality is of course a decision to be taken by future generations.

Very low-level waste (VLLW) is defined as waste that does not meet the criteria for exemption, but has such low activity content that it does not need a high level of containment and isolation. It is thus suitable for dis-



14.16 Disposal of very low-level waste at the disposal facility for very low-level waste at Morvilliers, France (© Emmanuel Gaffard).

posal in near-surface landfill-type facilities. An example of such a disposal facility is Morvilliers in France (Fig. 14.16). These types of facilities should also be part of the infrastructure needed in any individual country introducing nuclear power plants. They will be needed at the time of dismantling a power plant.

14.6 Conclusions

The management of radioactive waste has sometimes been seen as the Achilles heel of nuclear power production. It will require very long-term considerations also for the period after the nuclear power production has been stopped. Some of the radioactive wastes are very hazardous and will require very careful handling and management. They are also long-lived and will require isolation over hundreds to hundreds of thousands of years. The volumes to be handled are, however, quite small and the utmost care can be exercised without significantly increasing the cost of nuclear power production (a few percent of the production cost) or putting undue burdens on the future. All countries with nuclear power plants have an active programme to responsibly manage their radioactive waste by treatment, conditioning and storage today and by operating or developing disposal facilities for tomorrow. Preparations for disposal of low-level waste from reactor operation should be considered from an early phase of planning nuclear power as these types of waste will occur from the start of the reactor. They should be part of the infrastructure necessary for starting a nuclear power programme. Also the future handling of the spent nuclear fuel should be considered at an early stage, although construction of facilities will only be needed decades later.

In this chapter several examples of the management principles, strategies and methods have been described. It should be clear that the final choice of strategy will depend on the national conditions, e.g. size and prospects of the nuclear power programme, industrial capacity and geological conditions. It will often be too early for a country considering the introduction of nuclear power to decide from the beginning what strategies should be chosen, e.g. concerning reprocessing and recycling and concerning disposal. It is, however, very important that good comprehension of the options is developed early and to see how different options could be implemented in the specific country. It is also very important to ensure from the start of nuclear power production that funding will be available to take care of the waste (including decommissioning of the power plants) when needed, taking into account that many of these costs will appear long after the power production has stopped. It should be realized that introduction of nuclear power implies an undertaking for a hundred years or more.

14.7 References

IAEA (2006a), *Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management,* IAEA International Law Series No. 1, Vienna, IAEA; see also http://www-ns.iaea.org/conventions/

IAEA (2006b), *Fundamental Safety Principles,* IAEA Safety Standards Series No. SF-1, Vienna, IAEA

IAEA (2007a), *IAEA Safety Glossary – Terminology used in nuclear safety and radiation protection – 2007 Edition,* Vienna, IAEA

IAEA (2007b). *Milestones in the Development of a National Infrastructure for Nuclear Power*, IAEA Nuclear Energy Series No. NG-G-3.1, Vienna, IAEA

IAEA (2007c), *Selection of Away-from-Reactor Storage Facilities for Spent Fuel Storage*, IAEA TECDOC 1558, Vienna, IAEA

IAEA (2009a), *Classification of Radioactive Waste*, IAEA Safety Standards Series No. GSG-1,Vienna, IAEA

IAEA (2009b), *Policies and Strategies for Radioactive Waste Management*, IAEA Nuclear Energy Series No. NW-G-1.1, Vienna, IAEA

IAEA (2009c), *Regulations for the Safe Transport of Radioactive Material. 2009 edition,* IAEA Safety Standards Series No. TS-R-1, Vienna, IAEA

IAEA (2010), *International Project on Innovative Nuclear Reactors and Fuel Cycles. 2009 Progress Report*, Vienna, IAEA*;* see also http://www.iaea.org/INPRO/

NEA (1999a), *Progress Towards Geologic Disposal of Radioactive Waste. Where do we Stand?*, Paris, OECD

NEA (1999b), *Confidence in Long-Term Safety of Deep Geological Repositories. Its Development and Communication*, Paris, OECD

NEA (1999c), *Actinide and Fission Product Partitioning and Transmutation, Status and Assessment Report*, Paris, OECD

NEA (2009), *A Common Objective – A Variety of Paths*, Paris, OECD

Witherspoon P.A. and Bodvarsson G.S. (2006), *Geological Challenges in Radioactive Waste Isolation. Fourth Worldwide Review,* LBNL-59808, Berkeley, CA

15
The economics of nuclear power: past, present and future aspects

H.-H. ROGNER, International Atomic Energy Agency (IAEA), Austria

Abstract: The economics of nuclear power are reviewed from the perspectives of investments, finance and overall competitiveness versus alternatives on a life-cycle basis. Nuclear power plants, once built, are cheap to operate but their construction is expensive, and with current commercially available unit sizes they represent a sizable financial exposure and economic risks to investors. Long lead times for planning, licensing and construction as well as payback periods counted in decades further compound investor risks. Policies that reward environmental performance generally improve the competitiveness of nuclear power. The chapter touches upon direct economic costs as well as externalities and government policy in support of the technology.

Key words: nuclear power economics, finance, generating cost, externalities.

15.1 Introduction

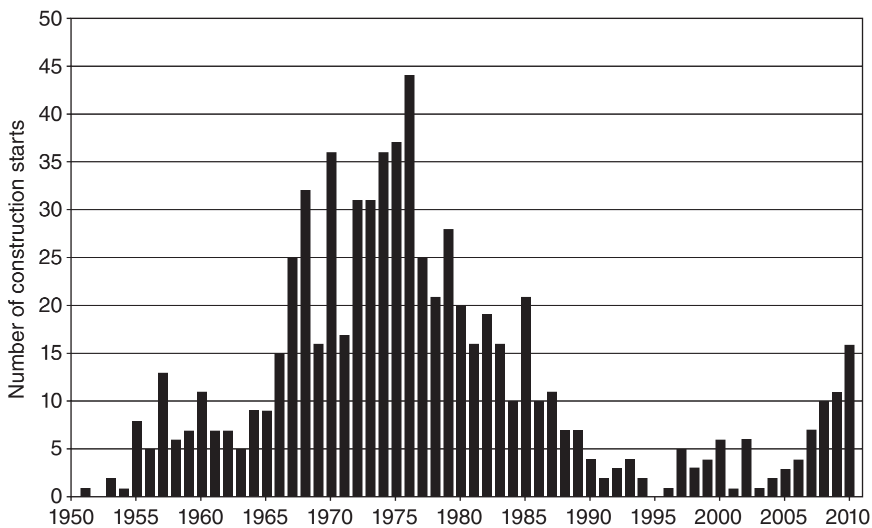
15.1.1 Current status of nuclear power in global energy supplies

By November 2011 there were 433 nuclear power plants in operation worldwide with a total installed generating capacity of 367 gigawatts (GWe). There were 65 plants under construction with a combined capacity of 62.6 GWe. In 2010 the global fleet of nuclear generating stations produced 2630 terawatt-hours (TWh) of electricity or about 13.5% of total supply.

Published by Woodhead Publishing Limited, 2012

The first decade of the twenty-first century was a paradoxical period for nuclear power. Projections of future growth were revised upwards year by year even though global installed nuclear generating capacity did not grow materially and actually declined after 2007 as several plants were retired and no new reactors were connected to the grid in 2008. It was the first year since 1955 without at least one new reactor coming on-line. There were, however, 10 construction starts, the most since 1987. In 2009 installed nuclear capacity dropped yet again, the first two-year drop in nuclear power's history, with three reactors being retired and only two new ones connected to the grid. But the projections for nuclear power growth by reputable international organizations were again revised upward, by about 8%, even as the world was still dealing with the financial and economic crises that started in late 2008. One reason for the higher projections was that construction starts on new reactors also increased. There were 11 new construction starts (see Fig. 15.1), extending a continuous upward trend that started in 2003. With 16 construction starts, the year 2010 witnessed a continuation of this trend and the 67 plants under construction at the end of 2010 is the highest number since 1987.

Then again, the share of nuclear-generated electricity in global supplies has been slipping throughout the twenty-first century. In 2009 it was just below the 14% mark (down from 16.8% in 2000) and many analysts have interpreted this as a clear sign of a nuclear demise as nuclear capacity growth was routinely outpaced by total capacity growth. Still, in 2010 every



15.1 Construction starts of nuclear power plants by year (IAEA, 2011).

seventh kilowatt-hour (kWh) produced globally is generated by nuclear power.

The short-term reality of declining market shares is in juxtaposition with the interest expressed by more than 60 countries currently without nuclear energy to add the technology to their national energy supply portfolio (IAEA, 2010).

15.1.2 Past (economic performance) experience – lesson learned

The promise of electricity 'too cheap to meter' of the 1950s and early 1960s brought about a quasi-unprecedented enthusiasm for a new technology throughout societies around the world. It was going to open up an era of abundant and clean electricity and stop the filth and smoke of oil- and coal-fired power plants. Numerous countries launched ambitious peaceful nuclear power programmes, a trend that was further fuelled by the oil supply crises and associated price hikes of 1973 and 1979. Global nuclear generating capacity expanded rapidly beginning in the 1960s from barely 1 GW to 325 GW by 1990. During these early years, nuclear power plants were enthusiastically supported and mostly funded by governments in large part to develop and commercialize this new technology. Utility orders began to mushroom by the 1970s and expectations were that nuclear power would provide the lion's share of electricity globally by the beginning of the twenty-first century. The order books of the nuclear industry were brimful as plant orders poured in by the hundreds.

Reality, however, proved different. Beginning in the early 1980s numerous orders were cancelled – even where plant construction had almost been completed – and global nuclear power faced stagnation which lasted until about 2002–03. There were many reasons for these cancellations and subsequent stagnation but the bottom line was economics. Given the current rising interest in nuclear power, a review of factors underlying past nuclear stagnation and whether the situation is different today than 30 years ago is in order. In essence, the following factors chiefly determined and will continue to determine the economics of nuclear power: market structure, government policy, generating costs relative to alternatives, finance, public acceptance and environmental performance.

Looking back to the 1980s, the nuclear stagnation is not attributable to a single factor but is rather the result of a combination of several (often unrelated) factors. Regarding demand, while the oil price shocks of the 1970s had been a major driver of the nuclear expansion, they also had prompted government policy mandating efficiency improvements throughout the energy system as well as the development of alternative energy sources. High fossil fuel prices not only provided incentives for accelerated exploration and development of non-OPEC oil resources but also resulted in structural economic change in many industrialized countries, i.e., a shift from energy-intensive primary and secondary manufacturing industries to tertiary (service and knowledge based) industries. The net effect of all these measures was a considerably lower growth in electricity demand which due to time lags became only evident by the early 1980s. Demand uncertainty – until then a relatively insignificant risk – became a new challenge. Risks were further compounded by the emergence of surplus generating capacity in many markets. Long lead times in power capacity planning and plant construction made it difficult to respond in a timely manner to the new demand situation. As a result many – nuclear and non-nuclear – power plant orders were cancelled or halted where construction was already underway.

At the time, electricity supply was viewed as a strategic good and most electricity utilities were government owned. In markets where utilities were privately held, strict regulatory oversight ensured cost controls, supply reliability and security. In either case, utilities were vertically integrated, viewed as natural monopolies without real market competition (except with other fuels). In return for quasi-guaranteed markets, utilities had a supply obligation at government-controlled sales revenues. Revenues were usually structured to cover actual fuel and variable operating and maintenance (O&M) costs plus, in the case of private sector ownership, a regulated (reasonable) return on capital. Through direct ownership or through the regulatory approval process, governments directly influenced investment decisions and technology choices.

In essence, privately owned utilities operated under a 'cost plus' scheme, i.e., they could essentially recoup all costs, including investments – even if these were higher than anticipated (unless imprudently incurred) or if demand turned out lower than projected. To that extent, the economic risk of electricity supply was entirely borne by the taxpayer. It was this low-risk framework that enabled utilities to invest in capital-intensive generating stations such as hydro or nuclear power. However, the situation changed in the 1980s, in large part in response to surplus generating capacities in many markets and the resulting widely differing rates between regional markets. Another change concerned a shift in the recognition of the different roles of public and private sector entities and their respective efficiency and effectiveness in decision making and risk management. Regulated electricity markets gave way to deregulation and market liberalization. Many government utilities were privatized. Electricity market competition, unbundling of generation, transmission and distribution instead of quasi-natural monopolies became the new paradigm in many countries for addressing surplus capacity and stranded costs, reducing rate differences between regional markets and encouraging electricity trade. Competition, partitioning and allocating risks to entities that are best positioned to manage them were hailed to improve efficiency and overall market transparency and ultimately incur lower costs to consumers. Clearly 'cost plus' rate setting as well as long-term investment planning and decision making became a thing of the past overtaken by short-term shareholder value optimization.

Investment in nuclear power, however, not only requires long-term planning but also involves long pay-back periods and lower returns than alternative investment opportunities – characteristics which proved incompatible with short-term shareholder optimization. The general retreat of government involvement (usually with longer planning horizons) in financing electricity sector investment – be it because of general divesture or the many other non-energy demands on government budgets or economic transition – further reduced the attractiveness and market potentials of nuclear power.

In addition, the track record of the nuclear industry to deliver nuclear power plants at budget and on schedule was marred as construction delays and cost overruns through the 1980s often became the norm rather than the exception. The plants built in the 1970s were scaled-up adaptations of smaller demonstration plants built in the 1960s, thus effectively representing a 'first-of-a-kind' experience. Often designs were being finalized on-the-fly during construction, resulting, at times, in widely differing final plant designs for initially identical units as different engineering approaches and design improvements provided for different solutions (NEA, 2009).

While extreme cases of cost overruns, e.g., of an order of magnitude, or delays of many years were rare, and many less extreme delays and overruns can be rationalized (see below), they brought many utilities to the brink of bankruptcy and the reputation of the industry with investors plummeted and has yet to be fully restored. Several factors – partly beyond the control of the industry – contributed to plant completion delays and cost overruns. The 1979 Three Mile Island (TMI) accident in the United States raised safety concerns and prompted regulators to toughen safety regulation. New regulatory requirements mandated upgrading of existing plants and plants under construction with additional and more complex safety features. For plants under construction this resulted in extended construction schedules and added costs; for completed plants it meant lengthy shutdowns and loss of sales revenues. Moreover, the early 1980s saw a period of two-digit interest rates and high inflation which further compounded cost overruns through cost escalation and accumulated interest during construction.

The TMI accident also adversely affected public and political acceptance and served as a wake-up call to investors about the economic and technical risks of nuclear power plants. Governments seeing their budgets stressed by cost overruns began to see the technology in a different light. The revised regulatory and plant licensing procedures also opened prospects for the involvement of civil society through public hearings, environmental impact assessments and legal intervention. Especially, anti-nuclear groups seized the opportunity and over time perfected the effectiveness of legal intervention, causing further delays and added costs.

As regards overall energy supply, the stepped-up investments in non-OPEC oil exploration and production capacity as well as the delayed effect of efficiency and performance standards began to impact the international oil market: lower demand was met by rising supplies exerting downward pressure on prices. This situation culminated in 1986 when OPEC lost control and global oil prices collapsed (and gas and coal prices followed suit), compromising the economic rationale for nuclear power. Plentiful cheap oil and gas on the one hand, and the advent of low capital cost, highly efficient combined cycle gas turbines (CCGT) with smaller unit sizes and considerably shorter construction and payback schedules than nuclear power (and coal), on the other hand, offered utilities less bulky and lower risk investment opportunities. Smaller unit sizes were highly welcome in markets with uncertain electricity demand prospects, and high returns were consistent with short-term profit and shareholder value maximization.

With plentiful cheap oil and gas available, energy supply security – the prime driver of nuclear power in the 1970s – was no longer a national policy concern in most countries. Environmental performance also appeared less a matter of concern. Policies targeted at controlling sulphur and nitrous oxide emissions chiefly responsible for local air pollution and regional acidification had taken effect already in many industrialized countries and the threat of climate change had not yet been high on the international environmental agenda.

In summary, the economics of the day had already disfavoured nuclear power with investors when the disastrous Chernobyl accident of April 1986 – like the straw that broke the camel's back – also turned the public at large against the technology. Many reactor orders not already cancelled for economic reasons were now stopped due to safety fears and several countries decided to abandon their national nuclear power programmes. The global nuclear power situation was further set back by the disintegration of the Soviet Union and consequent economic collapse.[1]

[1] This was the situation in Western Europe (except France where nuclear power construction continued well into the 1990s), North and Latin America but not necessarily in Asia (see below) where energy security remained high on the political agenda (due to lack of indigenous energy resources) or fast-growing and populous countries with ambitious economic development aspirations judged the benefits of nuclear higher than the risks. And in Latin America and Africa, nuclear power has played only a marginal role.

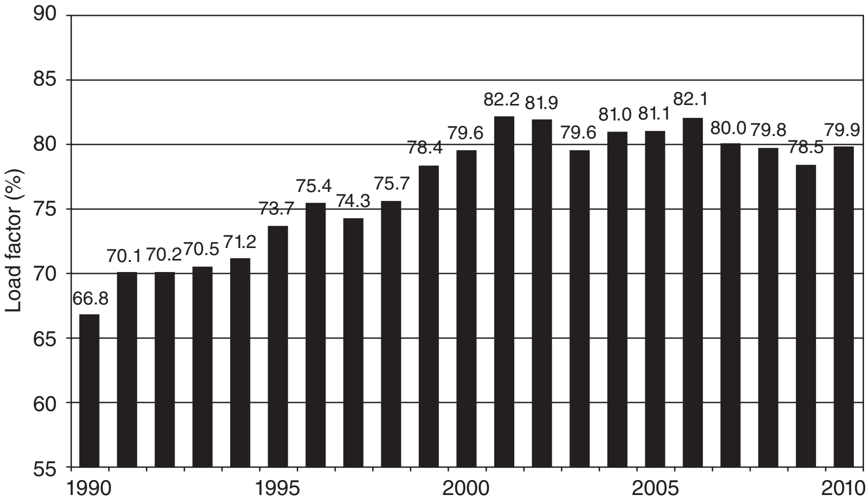
15.1.3 Impact on the nuclear industry

Few new reactors were ordered after 1986; the number coming on line from the mid-1980s little more than matched retirements. Chernobyl and electricity market liberalization were generally viewed as the final nails in the coffin of nuclear power. It was inconceivable for many analysts that nuclear power could survive in the absence of 'cost plus' pricing in a competitive market. While the number of construction starts indeed suggested an early demise of the industry, it was not lack of construction but market pressures and competition that forced it to streamline and consolidate operations. In a deregulated environment, with the rate base eliminated, revenues are based solely on the difference between a plant's operating and fuel (or short-run marginal) costs plus the remaining debt on yet to be depreciated assets and the market price of electricity.[2] Many analysts were of the opinion that the remaining debt on nuclear power plants would make them too expensive to compete with coal- and gas-fired generation. Economic rationale suggests, however, that an existing plant continues operating as long as revenues cover marginal operating cost irrespective of any debt as even small margins above short-run costs contribute to debt repayment and profits.

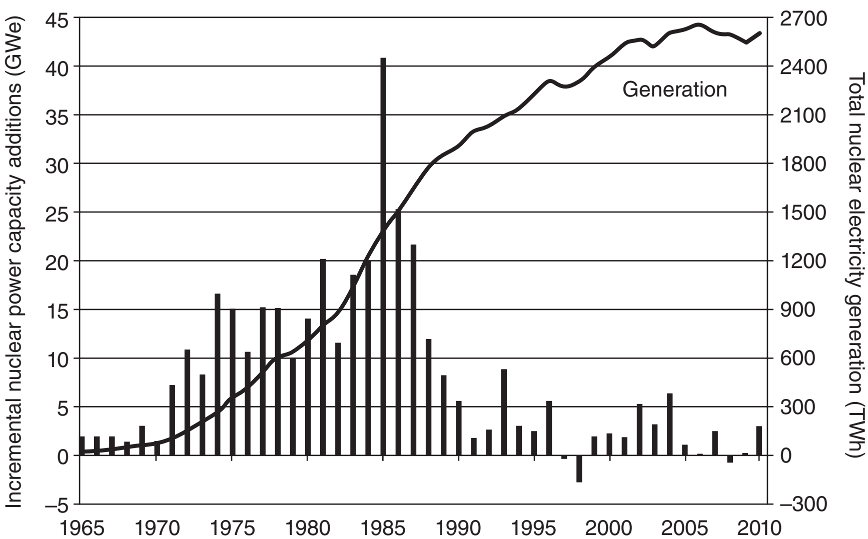
For a capital-intensive technology in a competitive market such as nuclear power, it is vital to put the assets to productive, i.e. revenue, generating use. The more kWh a plant generates, the lower its total production costs per kWh as fixed costs are distributed over more kWh. Until the early 1990s, the load factor (the percentage of time a plant generates full capacity electricity to the grid) of the global fleet of nuclear power plants hovered around 65%. Competition forced nuclear operators to condense maintenance outages, reduce overhead costs through consolidation of different plants, and implement numerous other management measures. By 2005 the global load factor reached more than 80% (see Fig. 15.2) which allowed continued growth in nuclear generation, despite aggregate capacity expanding only 14% over the period (see Figs 15.3 and 15.4). The vastly improved utilization of existing capacities worldwide corresponds to a virtual construction of more than 30 1000 MW nuclear power plants.

Variable operating costs, essentially fuel costs, are a comparative advantage of nuclear power, especially in competitive markets and when plants are fully depreciated, thus making licence extension highly profitable for many nuclear operators. The reason is straightforward: it costs considerably more to build any type of new generation – fossil, nuclear or renewable – than to invest in the maintenance/replacement of some nuclear components

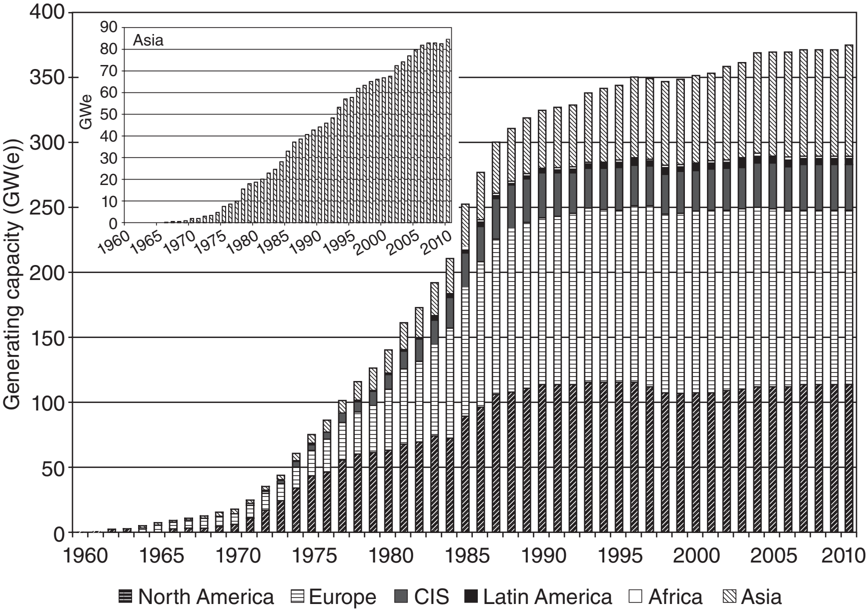
[2] For new plants, revenues must be adequate to cover total generation costs (long-run marginal costs), including the investment costs of the plant which in the case of nuclear are considerably higher than short-run marginal costs.



15.2 Development of the load factor of the global fleet of nuclear power plants (IAEA, 2011).



15.3 Annual incremental nuclear capacity additions and nuclear electricity generation (IAEA, 2011).



15.4 Global nuclear generating capacity in GW(e), 1960–2009 (IAEA, 2011).

and run a nuclear plant for an additional 20 years.[3] These investments usually also result in improved operating safety, power uprates and/or higher output (e.g. new turbines, more efficient steam generators), all of which further improve overall economics.

Another reason for the attractiveness of licence extension is public acceptance and greatly reduced licensing procedures compared with new build. As regards public acceptance, communities hosting nuclear power plants have had a positive decade-long experience living with the technology, i.e., a better comprehension of the associated benefits exceeding the risks.

15.1.4 Regional impacts differ

While capacity stagnation was the characteristic of nuclear power development in most world regions, this cannot be said about Asia (see Fig. 15.4). The surplus capacity situation did not exist really in the fast-growing industrialized countries with limited domestic energy resources (Japan or Republic of Korea) or in the even faster-growing populous developing

[3] Licence renewal for 20 years has been the practice in the USA. In other countries renewals are granted for different periods.

countries of China and India. Here energy security remained a high priority policy item and, in the cases of China and India, fuelling their economic development aspirations called for the development of all supply options. Moreover, electricity market liberalization was less pronounced in these countries than in North America or Europe and government involvement in energy system investment decisions (and finance) continued.

15.2 Economics today and tomorrow

15.2.1 What is new?

By the start of the twenty-first century, the background conditions for investing in new generating capacity had changed fundamentally. Fossil fuel prices increased dramatically (in large part by the accelerated demand in Asia, continued depletion of low-cost oil and gas occurrences and lack of investment in upstream operations) and fossil-sourced electricity no longer offered lower total generation costs in many markets. This improved not only the comparative economics of existing nuclear power plants (and spawned licence extensions) but also the prospects for new plant investment.

Energy security was back on the policy agenda of most countries, especially those with high energy import dependence. Nuclear power offers not only diversification, a cornerstone of energy security, but also relatively stable and predictable generating costs in the long run due to its small share of uranium costs in total generating costs. As well, uranium occurrences are more widely spread globally than fossil resources,[4] nuclear fuel volumes are small (and can be stored for several refuelling cycles) and refuelling schedules extend for as long as 18 to 24 months.

Next, climate change had become one of the most important energy and environmental policy challenges as manifested by the Kyoto Protocol (UN, 1998), the international environmental agreement under the United Nations Framework Convention on Climate Change (UNFCCC) aimed at the stabilization of atmospheric greenhouse gas (GHG) concentration at a level that would prevent dangerous anthropogenic interference with the climate system (UN, 1992). The Protocol was initially adopted in December 1997 in Kyoto, Japan, and entered into force in February 2005. On a life-cycle basis, nuclear power generates only a few grams of carbon dioxide (CO_2) per kWh – orders of magnitude lower than fossil fuels (in the absence of

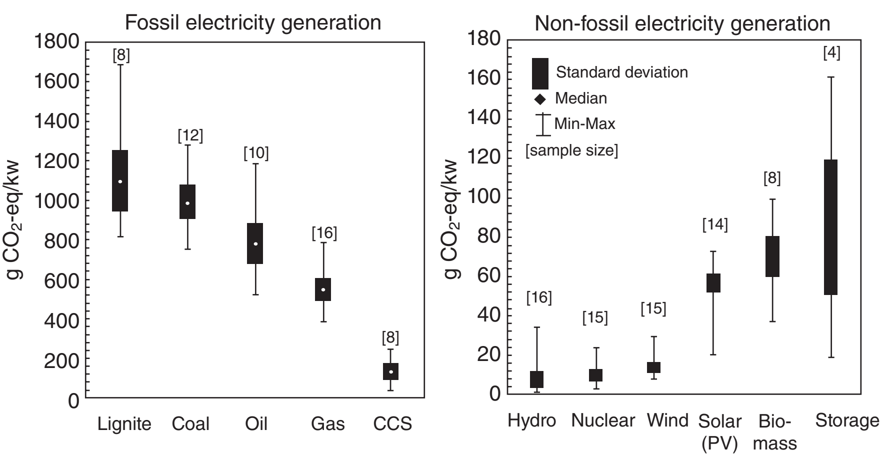
[4] Today's uranium production is dominated by seven to eight major producers – a result of a 15-year trough the global uranium market has faced until recently, a period characterized by shake-outs, mergers, consolidation and producers going out of business.

costly carbon dioxide capture and storage) – at least comparable with the emissions of the best performing renewable supply options (see Fig. 15.5).

15.2.2 Economic fundamentals

For investors and decision makers, it is the generating costs on a full life-cycle basis that ultimately matter. However, numerous direct and indirect factors determine these costs. Standard direct costs include investments, O&M and fuel costs. Indirect costs are overheads shared by several plants such as head office costs (billing, customer service, ancillary support services) but also external costs, i.e., costs inflicted on society at large that are not reflected in the price of electricity, thus not paid by the electricity generator. Typical external costs include, but are not limited to, the costs of air, water and land pollution from generation as well as fuel extraction and transport, accidents lacking sufficient liability coverage, and exposure to physical or economic disruption of supply lines.

The economics of a particular technology such as nuclear power cannot be analysed in isolation but only in comparison with its alternatives. For a private sector utility operating in a liberalized market the question usually is not either to generate or not to generate electricity but how to generate it most profitably. While nuclear energy is often very competitive on the basis of its low levelized life-cycle generating cost, its large upfront capital cost and long construction schedule make its financing more challenging compared to fossil fuel investments. Regulated utilities in quasi-monopolized markets or government-owned generators are bound by supply obligations. In both situations the do-nothing option does not exist

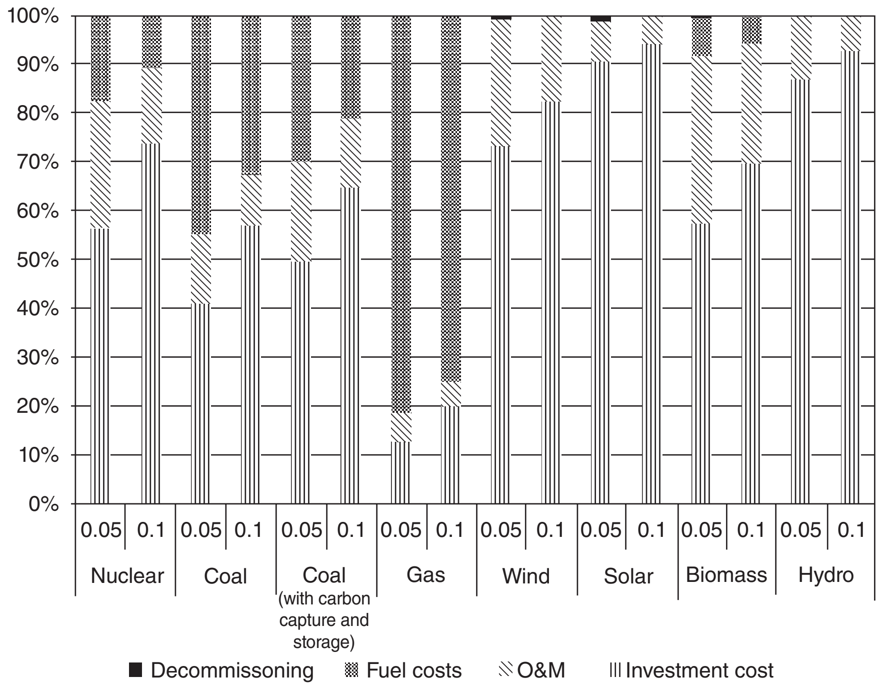


15.5 Life-cycle GHG emissions of selected electricity generating technologies. Adapted from Weisser (2007).

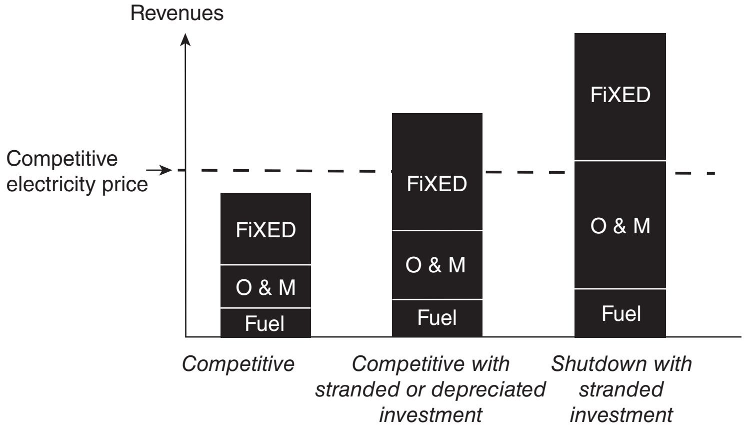
and rejecting one option, say nuclear power, requires the adoption of a non-nuclear alternative. Hence the actual economics of nuclear power can reasonably only be determined with regard to its alternatives (using a level playing field) under given market and local conditions.

Generic considerations

The generating cost structure of nuclear power is dominated by its capital costs which roughly account for 60% to 75% of total generating costs. This compares with about 25% to 40% for coal power plants without carbon capture and storage (CCS) and 12% to 18% for combine cycle gas technology (NEA/IEA, 2010). Figure 15.6 summarizes the relative shares of generating cost components for different generating options. Nuclear fuel costs assume a low share of 10% to less than 20% while the actual uranium share accounts for approximately 20% of fuel costs (or between 2% and 4% of total nuclear of generating costs). The small share of uranium in the generating cost structure makes total generating costs very predictable and stable in the long run. Even a 10-fold uranium price increase would increase nuclear's total costs by 18% to 32% depending on interest rates and



15.6 Generating cost structure of different electricity generating options (NEA and IEA, 2010).



15.7 Nuclear power plants in a competitive market. Fixed costs include undepreciated capital costs.

amortization periods. In contrast, for CCGT technology with a fuel cost share of about 70%, a mere doubling of natural gas prices translates into cost escalation of 73% to 80%. For CCGT, high fuel cost shares mean smaller margins over which the plant can make profits.

High capital cost is the single most important economic factor affecting the prospects for new nuclear build. Inherent uncertainty about electricity sales prices and the question whether revenues will be sufficient to cover full costs have become characteristics of liberalized electricity markets. Figure 15.7 depicts the revenue–cost positions typically found in competitive markets. Clearly, as long as revenues exceed total generating costs, plant operation is profitable. If prices drop below generating costs, high capital cost and low fuel cost technologies can still be competitive in the short run as long as revenues cover marginal operating costs and still allow for some contribution to debt service. But in the long run revenues must cover capital costs so that the operator can fully meet debt services and provide shareholders the expected return on investment. Once fully depreciated, its low fuel costs give nuclear a decisive edge over its competition. In contrast to nuclear power, CCGT has excellent load-following capability and thus can respond by reducing output during periods when the sales prices drop below short-run marginal costs. Or in cases where natural gas-fired CCGT is the lowest-cost provider and thus the price setter, utilities can pass higher gas fuel costs through to consumers, effectively allowing them to preserve their profit margins.[5]

The economics of nuclear power are not just an issue of competitive generating costs but also a matter of the wider economic and policy frame-

[5] Unless CCGT is no longer the lowest-cost generator.

work. This is particularly the case when the well-being of the public at large is at risk due to inadequate provision by markets – so-called externalities. Externalities relate to costs and benefits that are traditionally omitted from private sector evaluations of the economics of different generating options. Including these 'externalities' increases the likelihood of developing the most economical and sustainable power resource from a societal perspective (Roth and Ambs, 2004). Typical externalities are health and environmental damages caused by pollution from fossil fuel combustion, and energy security of limited liabilities of nuclear operators in case of accidents with off-site consequences. The associated damage costs are not borne by producers but by the public at large. Most importantly, the full costs of producing and using energy are underestimated by the markets and, therefore, not reflected in the market mechanisms determining the price of electricity. As a consequence, producers and consumers base their respective investment and purchase decisions on incorrect price signals (Bohi and Toman, 1996).

In the absence of government intervention, in each of the externality examples above, liberalized markets would fail to deliver an efficient or optimal resource allocation, resulting in loss of economic and social welfare.

For example, high fossil fuel import dependence, the prospects of price volatility, and technical or geopolitically motivated supply disruptions may adversely affect a country's energy security and any such incidence may result in a loss of social welfare. The external costs of the US oil import dependence have been estimated at \$3/gallon (Copulos, 2007). Consumers do not pay (or even know about) such costs, which, since they are not reflected in the market price, results in overconsumption. In essence, 'markets have no way of incorporating the energy security cost into the market transaction' (Tyner, 2007). Basic economic theory suggests a correction of such externalities through taxes, subsidies (for alternatives to oil), standards or some kind of regulation. Energy security considerations may prompt government policy to provide support for diversification of the country's energy mix. This could come in the form of either disincentives on the fuel/technology with high externalities, or attractive incentives for investment in lower externality options. For example, governments may support nuclear power projects (loan guarantees, power purchase agreements, direct involvement in the finance of the plant), despite it not being the least-cost supply option under standard direct cost accounting. The extra costs incurred by the policy can be interpreted as an insurance premium against the occurrence of the externality.

Similarly, policies targeted at mitigating climate change improve the economics of the low-carbon technology nuclear power. Even small cost adders (e.g., taxes or carbon prices under a cap-and-trade scheme) on CO_2 emissions can tilt the balance in favour of nuclear power (Rothwell, 2010;

KPMG, 2010). In many markets nuclear power could reduce CO_2 emissions at negative costs[6] (IPCC, 2007), especially those with a high dependence on imported coal or where nuclear power is excluded for political reasons.

15.2.3 Investment (capital) costs

Varying definitions and boundaries

The most important factor influencing the lifetime cost of a nuclear reactor is capital or construction cost. Investment in electricity generation is a multi-year affair and can take a decade or more from an early planning stage, conducting environmental impact assessments, obtaining construction permits, actual construction and plant commissioning before the plant produces the first kWh. Necessarily the actual financial outlays are spread over this period.

Despite its crucial role in determining the economics of nuclear power, investment costs of nuclear power plants remain a mysterious affair. The fact that the investment in a nuclear power project encompasses numerous, often site- or project-specific components ranging from site acquisition and preparation to bid evaluation, construction, licensing and grid integration, mandates unambiguously defined project boundaries, i.e., what is included in an investment cost quotation and what is not. It also requires clarity about the cost of finance during construction, currency exchange rates used, inflation over the construction periods, taxes or subsidies. Otherwise cost comparisons are meaningless.

At the most aggregate level, total investment costs equal 'overnight costs' (OC) plus interest during construction (IDC). The term OC is often used to express what the investment of a project would cost if it were built 'overnight', i.e., as if money had no time value.

IDC are the financing costs for plant construction until the plant is connected to the grid and generates revenues. Because it can take as much as 10 years or more to bring a nuclear power plant from planning to completion, IDC alone can tilt the balance between an economically viable or unviable project. Their long construction periods and high up-front investment requirements make nuclear power projects very sensitive to IDC, and thus to construction delays.

Overnight costs

The principal components of OC are engineering–procurement–construction (EPC) costs, owner's costs and contingency costs. EPC represent the bare costs of plant construction comprising direct (equipment,

[6] The impact of carbon penalties is calculated on a LCOE basis and therefore ignores barriers such as access to capital.

material, labour) and indirect (engineering/construction services) components. Under an EPC contract, the contractor – usually the vendor – is responsible for the engineering design, including adaptation to match site and other location-specific conditions, production or procurement of the necessary plant components as well as materials, and plant construction. Given the complexity of a nuclear power plant and its financial dimensions (economic risks), the contractor usually subcontracts parts of the work or shares parts with the plant owner or both. In essence, subcontracting and owner involvement are a measure of risk management (plant completion risk).

Owner's costs are additional investment expenditures borne by the plant owner and usually relate to costs associated with property (land) acquisition, site selection and preparation, bid evaluation, cooling infrastructure, administration and associated buildings, site works, project management, permits, legal services, licences, local taxes, staff and operator training, and possibly also expenditures for connecting the plant to the grid, i.e., switchyards and transmission infrastructure.

Contingencies are provisions for any unforeseen or unplanned expenditures associated with the project. They are generally estimated as a specified percentage of EPC but also depend on the type of contract arrangement (turnkey contract or several contracts managed by the plant owner or cost-plus contracts).

The absolute and relative values of these OC components depend on location and plant design and therefore can vary considerably even within a country and for plants of similar design and size. The major factors in variability across countries include domestic labour and material costs, site-specific conditions and readily available infrastructures, finance arrangements and interest rates, institutional and regulatory framework, standardization and multiple-plant versus single-plant construction (economies of scale). They are also a function of the localization rate, i.e. the ratio between imported and locally manufactured or procured components and participation in the civil works. The availability of nuclear-specific skilled tradespeople and engineering capability is another factor affecting OCs. Site- and geography-specific conditions may add costs to an otherwise standard design, such as additional design and engineering costs for measures to protect a plant in an earthquake-prone location.

Unit size and plant design are other factors that explain OC cost differences. Typically, smaller plants have higher specific investment costs (i.e. dollars per kW(e)) than larger plants, since certain cost components are relatively independent of size. For example, Westinghouse's AP-1000 design is 80% more powerful than its AP-600 design, but the AP-1000's overnight costs are only 15% to 18% higher than the AP-600's (RWE Nukem, 2002).

Regulatory intervention can add to overnight costs, especially if it requires design modifications once the project is well underway. The exact impact of regulatory changes on cost is elusive because the regulatory process varies across regions. A number of studies have tried to quantify the impact of regulation on nuclear power investment costs but have not generated broadly applicable quantitative results beyond the straightforward reality that construction delays increase investment costs (Mooz, 1979; Paik and Schriver, 1979; Komanoff, 1981; Zimmerman, 1982; Cantor and Hewlett, 1988; McCabe, 1996; Canterbery *et al.*, 1996).

For new designs, or for construction in new environments, OC may include first-of-a-kind (FOAK) costs. FOAK costs include a particularly high share of contingency costs to cover unforeseen events given the lack of experience with the design, the environment or the country. They can add as much as 35% to OC (UoC, 2004). Costs are lower for subsequent units, but some (decreasing) additional costs will persist until experience has been accumulated on several (about five to eight) essentially identical designs. For example, Progress Energy recently announced overnight costs of $3376/kW(e) for a second AP-1000 at its Levy County site, substantially lower than the first unit's $5144/kW(e). And the Russian Federation's Kaliningrad-2 cost $1667/kW(e), half the cost of Kaliningrad-1. In these examples the cost reductions also reflect the facts that some site preparation costs incurred for the first unit are not reincurred for the second unit and the vendors' allocations of costs among the two units are to some extent arbitrary.

The specific components of FOAK costs are uncertain and prone to escalation. For example, the OC cost estimate for Olkiluto-3, a FOAK third-generation European Pressurized Reactor (EPR), has reportedly risen from €3.0 billion to €5.3 billion due to construction delays caused by FOAK-related quality issues, design revisions and approvals, and logistic challenges not experienced for a long time (NW, 2010a; KPMG, 2010).

International comparisons of investment costs are also often obscured by the unavailability of information about the exchange rates[7] that are used and, if escalation costs are included, about the components that are affected and the escalation rates assumed. Finally, OC may include the initial core load of nuclear fuel.

The percentage of each OC cost component varies according to several studies that include both data for plants that have been built and estimates for future plants (Kozlov, 2004; UoC, 2004; Scoggs, 2007). For example, EPC ranges from 73% to 97%, owner's cost between 2% and 15% and contingencies between 1% and 13% of total OC. The different percentages reflect

[7] This matters more when exchange rates are more volatile, as they have been recently. For example, in July 2008, €1 = $1.60; in July 2010, €1 = $1.29.

various cost-shaping factors such as plant design, whether it is built on an existing or greenfield site, economies of scale (in terms of both unit size and the number of units previously built), contractual arrangements and the cost of labour. For example, low owner's costs may indicate a project built on an existing site or a local government subsidy for site development and preparation. Low contingency costs might indicate a turnkey contract,[8] while high contingency costs might indicate a cost-plus EPC contract.

Interest during construction (IDC)

While OC are important for vendors for preparing their cost calculations and bids, it is the sum of OC and IDC that utilities must arrange financing for. The investment decision, however, is usually guided by a comparison of the total estimated generating costs, i.e. OC plus IDC plus estimated future fuel, operating and maintenance costs, of nuclear power to the same sum for alternative electricity generating options.

Four factors determine the IDC of a construction project: (a) OC, (b) the construction period, (c) the distribution of the OC over the construction period, and (d) the return on equity to shareholders and the interest rate, or rates, to be paid for loans (different rates may apply to different plant components or construction stages). IDC adds an extra layer of uncertainty to the final investment costs of a nuclear power plant. While interest rates can usually be fixed before construction begins or, if they are variable, hedged through various financial instruments, the largest uncertainty in IDC arises from possible construction delays. When cost overruns occur, the largest portion is generally due to construction delays. To the extent that regulatory intervention during plant construction causes delays, it will also increase IDC.

Table 15.1 shows that an increase in the construction period from four years to six or 10 years can increase IDC's share of total investment costs

[8] Turnkey contracting – sometimes also referred to as 'Lump Sum Turnkey' or 'LSTK' – allocates all responsibilities to a contractor to design, build and deliver the project on time and to a required performance level, in return for payment of a fixed price (lump sum). A key feature of the turnkey approach is the requirement for the contractor to prove the reliability and performance of the plant and equipment. A lump sum turnkey price will include contingency allowances to hedge against the risk of things costing more or taking longer to deliver (Hosie, 2007). Because the plant completion risk rests fully with the contractor, turnkey contracts come with a premium and thus are more expensive than if the owner is also the architect–engineer and builds the plant on a 'cost plus' basis. Turnkey projects are popular in project-financed deals, where lenders require greater certainty about a project's final costs than is allowed for under contracts that reflect the traditional allocation of risks (Hosie, 2007).

Table 15.1 Construction duration and IDC share in total investment costs based on OC of $2000 per kW installed and a uniform distribution of OC over the construction period

Construction period	Interest rate (%)	IDC ($)	Total cost ($)	IDC share of plant cost (%)
4 years	10	553	2553	28
6 years	10	929	2929	41
10 years	10	1506	3506	75

from 28% to 41% or 75%, assuming OC of $2000/kW(e), a uniform distribution of OC over the construction period and a 10% real interest rate. If the interest rate were only 5%, IDC would be 13%, 19% and 32%, respectively, of total investment costs.

15.2.4 Operating and maintenance costs

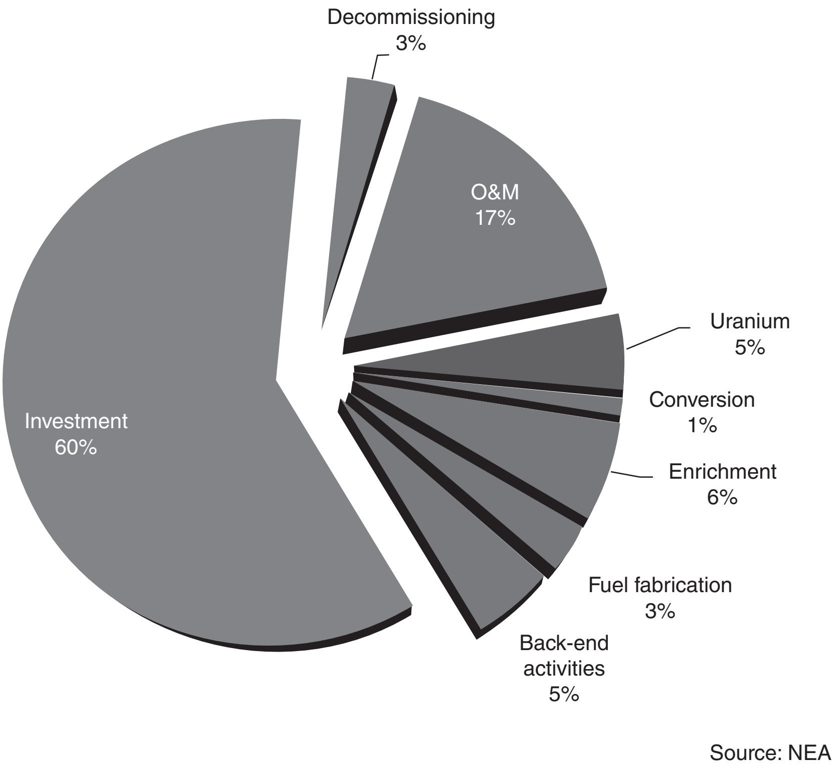
O&M costs of nuclear power plants are the non-fuel cycle costs for plant operation and related services and are generally divided into fixed (independent of electricity generation) and variable cost components. Essential O&M cost components are salaries for plant staffing and costs for materials, liability insurance, decommissioning, security, outsourced support services, administration and maintenance.

The O&M costs are further determined by the size and type of plant and the mode of operation (load-following or base-load operation). The number of similar units at a particular site has a strong influence on the O&M cost components.

15.2.5 Fuel costs

The term nuclear fuel costs often refers to nuclear fuel cycle costs which in many cases includes the costs for the front end and back end of the fuel cycle. The front-end or fuel input costs of the nuclear fuel cycle are determined by the prices of uranium mining and milling, conversion to UF_6, enrichment, if applicable, fuel assembly fabrication and interest on fuel in inventory. Back-end costs include those for reprocessing, if applicable, and disposal of high-level radioactive waste or spent fuel and for plant decommissioning (after final closure of the plant) and site rehabilitation.

Historically, nuclear fuel costs have varied between 10% and 20% of total generating cost (see Fig. 15.8) depending on prevailing uranium resource and enrichment costs, interest rates and whether or not back-end costs are included in fuel costs or treated as part of the variable O&M costs. Although generating costs are location- and design-specific, Fig. 15.8 indicates the relative shares of the cost components of nuclear electricity generation.



15.8 Nuclear power life-cycle generating costs (NEA, 2003). Fuel costs for nuclear comprise the costs of the full nuclear fuel cycle including spent fuel reprocessing or disposal.

Uranium metal and the price of enrichment services are the cost components most susceptible to fluctuation and supply and demand imbalances.

Uranium market and nuclear fuel cycle considerations

Uranium prices have been volatile over the past 30 years. The end of the Cold War curtailed the need for large stockpiles of military fissile materials, and the bleak prospect for civilian nuclear power during the 1990s enticed utilities to reduce their uranium inventories. So-called secondary uranium sources (reactor fuel derived from warheads, military and commercial inventories, re-enrichment of depleted uranium tails, as well as enriching at lower tail assays, reprocessed uranium and mixed oxide fuel) became increasingly available, e.g. through the 1993 agreement between the United States and the Russian Federation to convert highly enriched uranium (HEU) from nuclear warheads into low-enriched uranium for reactor fuel (also known as the Megatonnes to Megawatt programme). Low-cost secondary sources penetrating the uranium market and a general perception during the 1990s that nuclear power is a technology inevitably in decline suppressed uranium prices and mine production. Ever since 1990 annual fresh uranium production has fallen short of annual reactor requirements. Historically, low spot market prices threatened economic survival of many mines. Without clear long-term demand signals from the marketplace, the uranium industry has been reluctant to invest in new mine capacities or to pursue large-scale uranium exploration. Meanwhile, global production had progressively declined to less than 60% of reactor requirements. Clearly, uranium prices no longer reflected longer-term production capacities (Rogner, 2007).

Shortly after prices hit the historical low, a series of events uncovered the long-ignored demand/supply imbalance and caused prices to rise. On the demand side, since 1990 rising plant factors of the world's nuclear fleet added incrementally to annual reactor fuel requirements the equivalent of more than 30 GWe. A series of licence renewals for existing reactors that began around the turn of the century sent plant operators out to secure fuel for another 20 years or so. Another change was the growth of nuclear power in the developing economies of China and India, countries that had either not participated in the market to a great extent or not participated at all. While demand was picking up momentum, supply from mine output continued to be underprovided. In fact, in the face of rising demand several technical mishaps at major production centres reduced global mine output and prices began to rise. Moreover, the longer-term availability of secondary sources from military arsenals is politically determined and thus uncertain and the bulk of future uranium supply had to be provided by additional mine output, i.e., investment in exploration and development of new mines and mills. Given lead times of 5–10 years for new mining capacity to come on-line, in the short run production cannot increase rapidly despite rising demand. Beginning in 2004, the general demand-driven price acceleration of fossil fuels, materials and commodities further aggravated uranium prices and, by 2007, spot prices had exploded almost 20-fold.

As for almost all commodities, uranium market conditions abruptly changed with the onset of the financial and economic crises in 2008. At the close of 2009 spot prices were about 35% below their mid-2007 peak of \$350/kg U. Yet compared with other commodities, the uranium market weathered the storm fairly well. Uranium is generally better protected against aberrations than other markets. For one, short-run reactor uranium requirements are relatively stable as existing nuclear power plants are usually the lowest-cost generators on the grid and global annual reactor requirements of uranium of approximately 67,000 U remained unchanged. For another, most uranium (about 85%) is supplied under long-term contracts, where the pricing is shielded from sudden market fluctuations. New contracts or contract renewals then tend to also reflect the current spot price situation among other demand and supply factors. Typically, average long-term multiannual contract prices have been about half the going spot market price.

What brought down spot prices – in addition to the precipitous fall of energy, material and commodity prices – were those hedge funds and investors who since 2004 have traded in uranium and who, to a certain extent, added fuel to the 2004–08 spot price rally and, as a result of the financial crisis, were forced to sell their uranium positions due to cash requirements.

The longer-run price outlook, however, depends on whether or not above-ground investment in exploration and mining capacity will be forthcoming and mobilize the below-ground uranium resources. While global uranium resources are plentiful (NEA, 2010; Rogner, 2010) and the recent prices have stimulated both exploration and investment in new mining capacity, it remains to be seen if these are sufficient to meet additional demand caused by the expected nuclear renaissance but also to compensate for the likely decline in availability of secondary sources. Therefore, considerable uncertainty about future uranium prices remains. In the long run, uranium prices will be capped by the possibility of reprocessing of spent fuel. Except in Japan, no new commercial reprocessing facilities have been built for decades. The existing quasi-commercially operating plants in France and the United Kingdom initially served military purposes and were adapted or rebuilt for spent fuel reprocessing in the 1960s and 1970s under fundamentally different conditions (e.g., exponential growth of nuclear power, perceived limited uranium availability, continued demands for military purposes) and expectations of future nuclear power development in which plutonium-fuelled fast breeder reactors played a central role. This future did not materialize, but reprocessing continued, often rationalized as an integral part of a nation's nuclear waste management strategy or as a source for mixed oxide fuel (MOX) production and reuse in standard light water reactors (LWR). In any case, the expensive construction costs were quasi-stranded (sunk costs) and reprocessing services were offered internationally at attractive terms. In short, the economics of reprocessing in the near future hinge upon substantially higher uranium prices (or the equivalent of the revival of fast breeder reactor technology). During the last decade several studies attempted to cut through the complexity of reprocessing with its capital and operating cost depending on a mix of potential credits for recovered fissile materials, different waste volumes, interim storage requirements, high-level waste treatment and final disposal, and to determine break-even points with regard to uranium costs and once-through fuel cycles. For example, Bunn *et al.* (2003) concluded that 'at a central reprocessing price of $1000/kg of heavy metal (kgHM), and with other central estimates for the key fuel cycle parameters, reprocessing and recycling plutonium in existing light-water reactors (LWRs) will be more expensive than direct disposal of spent fuel until the uranium price reaches over $360/kg of uranium metal.' Likewise, the study *The Future of Nuclear Power* (Deutch and Moniz, 2003) concluded similarly, and that conclusion was repeated in the authors' 2009 update (Deutch *et al.*, 2009) which stated that 'given the assumptions about uranium resource availability and new plant deployment rates, the cost of recycle is unfavorable compared to a once-through cycle, but the cost differential is small relative to the total cost of nuclear power generation'.

The crux of the matter of all things concerning the nuclear fuel cycle is contained in the last part of the conclusion: nuclear fuel cycle costs have been and will continue to be a small cost component in total nuclear generating costs. The actual fuel costs per MWh are a function of the front-end costs, capacity factor and burn-up (number of MWh per unit of mass generated from the fuel) and the overall spent fuel management strategy (once-through or reprocessing and reuse). A very recent study estimated the once-through fuel cycle cost for LWRs at $8.67/MWh or some 10% to 14% of total generating costs (Rothwell, 2010).

The cost components for spent fuel management, disposal and decommissioning are accumulated in escrow funds (or equivalent schemes) as the plant operates and account for approximately 10% of total O&M costs (or approximately $1/MWh). However, these components can vary widely depending on reactor technology, regulatory requirements and the time frame over which these must be accumulated.

The lifetime fuel requirements (in terms of volume) of nuclear power plants are relatively small (compared with fossil generation) and so are the amounts of spent fuel and waste. But spent fuel is radioactive and must be kept isolated from the environment. Most countries require spent fuel to be stored at the plant site for an interim period until its radioactive inventory is greatly reduced and the fuel can eventually be transferred to a permanent repository outside the plant site. If spent fuel is accumulated over many years or the entire plant life, sufficient storage capacity must be provided.

Cost escalation

Investment costs for all power plants began to ascend quite steeply around 2005 and by 2008 had more than doubled for conventional coal technology and especially for nuclear power. This sharp increase coincided with the rapid increase in world market prices of energy and materials (e.g. cement and the full spectrum of metals). While these price hikes have clearly been one element pushing investment costs, they alone do not explain their magnitude. They are rather the result of a combination of several coinciding factors such as an above-average demand for generating capacity in Asia, an ageing fleet of power plants in North America and Europe requiring replacement or refurbishment for environmental reasons, as well as efficiency improvements due to high fuel prices and a global power equipment manufacturing industry characterized by relatively minimal expansion for over a decade – hence little spare manufacturing capacity.

Regarding nuclear power, globally only a few manufacturers were capable of producing heavy forging equipment such as reactor pressure vessels and steam generators. By 2008 lead times of 50 months and more had become commonplace. Backlogs started to accumulate with the licence extensions for existing reactors which often require replacing steam generators and other heavy components. The rising interest in new nuclear build and the accompanying pre-orders further added to the backlog. Full order books allow manufacturers to command higher margins and thus exert upward pressures on prices.

By 2007–08 prices for new nuclear build announced by utilities that are expected to deliver the first kWh to the grid sometime during 2017–20 started to ascend steeply, deviating considerably from the previous $1000 to $2500 per kW range (NEA and IEA, 2005). For example, in October 2007 Florida Power & Light released projected investment costs of $12.1 billion to $17.8 billion for two new Westinghouse AP1000 reactors (1100 MW each) or $5500 to $8100 per kWe at its proposed Turkey Point site. In March 2008 Progress Energy announced that its two new AP1000 units on a greenfield site in Florida would cost it about $14 billion or some $6360 per kW(e). In November 2009 Citigroup Investment research put construction costs for new nuclear build in the United Kingdom in the range of $3700 to $5200 per kW(e).

In May 2010, Progress Energy raised the estimated cost of its proposed 1100 MW reactors at the Levy nuclear power plant in Florida from $17.2 billion to $22.5 billion and delayed its start-up to 2021 due to a delay in licensing the reactors (Reuters, 2010).

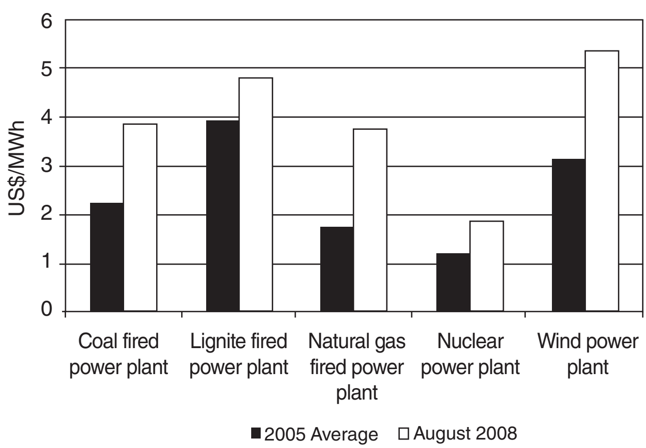
In contrast, the US Congressional Budget Office (CBO, 2008) quotes $2300 per kW(e) for a generic design in a report published in 2008 which is in line with NRG's August 2007 estimated cost range for two 1350 MW advanced boiling water reactors (ABWR) to be built in South Texas of $2200 to $2600 per kW(e).

The first two out of a total of six domestically developed 1000 MWe CPR-1000 pressurized water reactors in China are quoted at a cost of $1850 per kW (WNN, 2010). The first unit is scheduled to begin operating in 2015, followed by the second unit in 2016. Some 87% of the equipment to be used is being provided by Chinese suppliers (WNN, 2010).

Common to all these cost quotes is that they do not convey what is included and what is not. In essence these quotes are not comparable, although in the public mind they are all considered real. Clearly, this divergence of investment costs causes confusion and, taken at face value, seriously questions the economics of new nuclear build.

The exclusive focus on nuclear investment costs as a single data point ignores the effects of the material price hikes, the manufacturing constraints for power equipment and the shortage of skilled labour on all electricity generating technologies. Given the favourable material intensity per MWh of electricity generated (on a lifetime basis) from nuclear power compared with fossil and renewable alternatives, the energy and material price hikes have affected NPP costs less than the alternatives. Figure 15.9 compares the material-related change in generating costs for new power projects between 2005 and 2008 just before the onset of the financial crisis (ENEF, 2010). While all generating options saw steep increases in material-related costs, nuclear power was least affected.

But the capital cost quotations by utilities suggest a different picture, so which other factors may explain the recent enormous escalation of nuclear investment costs? At least four potential causes have been already identified: (a) varying definitions of investment costs, (b) boundaries of the analysis, (c) interest rates and market structures, and (d) price expectations (inflation) for materials, equipment and labour. The next paragraphs attempt to put the above divergent cost quotations into perspective with the help of a brief numerical example.



15.9 Material-related price jumps for new power projects (ENEF, 2010).

The compounding impact of interest and inflation

Table 15.2 provides cost data on the construction of a hypothetical nuclear power plant, and lays out a few commonly used but very different methods for quoting these same costs. The example demonstrates a large disparity of quotations, even when the underlying plant and cost data are the same.

In the example, construction is planned to occur over a five-year period running from 2014 through 2018, so that the plant is ready for grid connection at the start of 2019. The utility receives an offer from a vendor for the construction of the nuclear island and turbine-generator unit under an EPC contract. Rows [3] and [4] of Table 15.2 show a typical construction cost schedule. The vendor's total EPC OC quoted in 2010 prices and exchange rates is $3500 per kW(e) or $3.5 billion for a 1000 MW NPP.

Lines [5]–[11] show how the cost for the same plant is typically quoted by a utility as it seeks approval for the plant (in regulated markets) or finance and equity partners (in deregulated markets). Line [5] is the vendor's quoted EPC OC cost, but these figures have been adjusted for inflation (here assumed to be 3% per year) so that each year's figure reflects the expected nominal expenditure in that year. Line [6] shows the owner's costs, i.e., the balance-of-plant, contingency and other costs that the utility has to cover out of its own pocket, in addition to the vendor EPC costs. The owner's costs shown in line [6] are taken at 20% of the EPC OC figures of line [5].

Line [7] shows the cost of transmission system upgrades which are assumed to be necessary to deliver electricity to the grid once the hypothetical plant is completed at the end of 2018. Line [8] is the sum of lines [5], [6] and [7]. This total cost, which is exclusive of IDC, amounts to $5520 per kW(e) and serves as the basis for the IDC calculations in line [9] assuming a weighted average cost of capital (WACC) of 12%.

Line [10] shows the total costs as expended, inclusive of IDC. Line [11] accumulates total annual costs. By the end of 2018, when plant construction is completed and ready for grid connection, this total cost, inclusive of IDC, is $6930 per kW(e).[9] This is almost twice the vendor's EPC OC of $3500 per kW(e). The difference between the two estimates is merely a question of the method of quotation, i.e., of what is in and what is out and how the dollar expenditures are denominated, whether in 2010 dollars or in nominal dollars (dollars as expended). If expressed in dollars at 2010 prices, i.e., what the plant would cost at the planning stage, the nominal total of $6930 corresponds to $5752 per kW(e). In contrast, if expressed in 2019

[9] Even slightly higher cost escalation rates feared by many utilities in 2008 in the face of skyrocketing material and commodity prices lead to substantially higher costs in nominal terms.

Table 15.2 Alternative cost quotation methods for nuclear power plants illustrated with a hypothetical example

		[A]	[B]	[C]	[D]	[E]	[F]
[1]	Year	2014	2015	2016	2017	2018	Total
[2]	Construction period (relative to grid connection)	–4	–3	–2	–1	0	
[3]	Construction schedule as a fraction of OCs, $2010	10%	25%	31%	25%	10%	100%
[4]	Vendor overnight cost, $2010	334	875	1082	875	334	3500
[5]	Vendor OC, nominal $ as expended at 3% inflation and cost escalation	376	1014	1292	1076	423	4181
[6]	Owner's costs including contingency, nominal $ as expended	75	203	258	215	85	836
[7]	Transmission system upgrades, nominal $ as expended				145	57	202
[8]	Total cost excluding IDC, nominal $ as expended	451	1217	1550	1436	565	5220
[9]	IDC at 12%	27	130	312	529	712	1711
[10]	Total cost including IDC, nominal $ as expended	478	1348	1862	1965	1277	6930
[11]	Cumulative total cost including IDC	478	1826	3688	5653	6930	
[12]	Total bus bar cost, nominal $ as expended	478	1348	1862	1965	1277	6930
[13]	Total bus bar cost, $2010	425	1163	1560	1598	1008	5753
[14]	Total cost including IDC, $2010	425	1163	1560	1598	1008	5753
[15]	Total cost including IDC, $2018	538	1473	1976	2024	1277	7288

Notes: All figures in $/kW. Example assumes a total EPC overnight cost of $3500, an inflation rate of 3% per year, a 20% factor for owner's cost and an allowed capital recovery charge (or WACC) of 12%.

Rows [1]–[15], columns [A]–[E]:

[3] Rate of expenditures is given.

[4] = [3] $*$ $3500

[5] = [4] $* (1.03)^2 - 2010$

[6] = 20% $*$ [5]

[7] Transmission expenditures are given

[8] = [5] + [6] + [7]

[9] [9B] = [11A] $*$ 12% + 0.5 $*$ [8B] $*$ 12% and so on.

[10] = [8] + [9]

[11] [11B] = [11A] + [8B] + [9B]

[12] = [10]

[13] = [12] $* (1.12)^{(2010-[1])}$

[14] = [12] $* (1.03)^{(2010-[1])}$

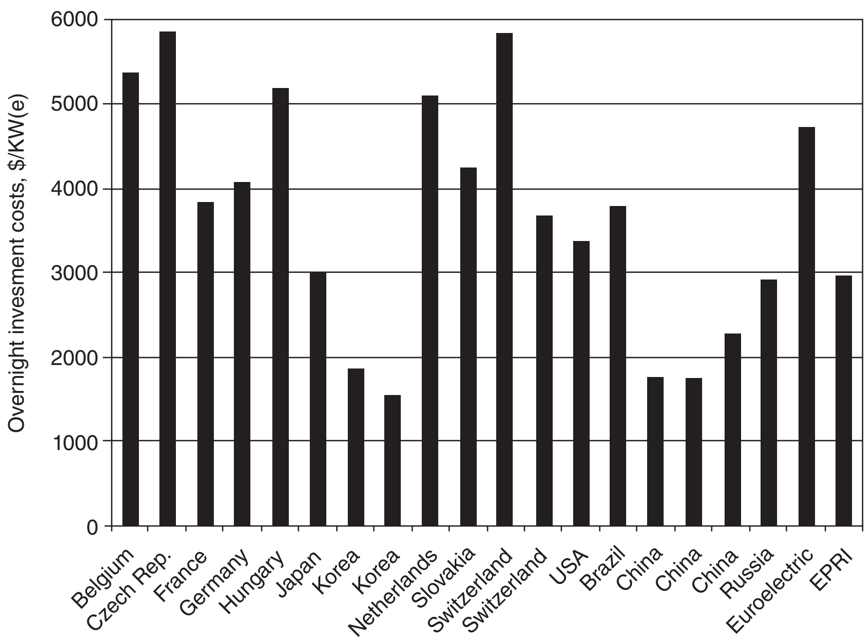
[15] = [12] $* (1.03)^{(2018-[1])}$

Source: adapted from Du and Parsons (2009).

prices, i.e., when the plant starts generating revenue, the cost is $7288 per kW(e).

A structured approach as outlined above allows a better comparison of cost estimates from different sources. However, it requires additional information than is commonly contained in media reports, utility or government announcements. Irrespective of the level of information available, it can help to identify inconsistencies in the quotations or define the set of common cost components upon which consistent comparisons can be made.

An even more transparent approach is the use of harmonized boundaries and assumptions. This not only facilitates comparisons of the costs of different nuclear power projects but also compares nuclear power with alternatives. The recent OECD report *Projected Costs of Generating Electricity* (NEA and IEA, 2010) followed the harmonized approach. Despite the harmonization, the report presents nuclear OC between $1560/kW(e) and $5860/kW(e) – a much wider range than five years ago – which shows continued uncertainty about nuclear power OC. Altogether 14 countries, all of which operate nuclear power plants, and two industrial associations contributed data for a total of 20 prospective nuclear projects (see Fig. 15.10). At the lower end of the OC estimates are China, Japan, Korea and Russia, i.e. countries with ongoing construction experience. At the higher end, OC



15.10 Expected overnight cost of nuclear power plants (NEA and IEA, 2010).

often reflect FOAK costs – either truly for the first construction of a design never built before (e.g., the EPR at Olkiluoto in Finland), for construction in a region or country without nuclear power (e.g., UAE or Vietnam) or for new construction in countries where active nuclear power construction stopped decades ago (e.g., USA, Belgium, Switzerland or UK).

15.2.6 Finance[10]

The basics

Equity and debt are the basic elements of capital finance. Equity finance means taking ownership, i.e., raising capital by selling shares of ownership in a venture. Sponsors may buy shares themselves (internal equity) or sell shares (external equity). Equity owners are attracted by the potential for profit (from electricity sales) compared to other investment opportunities. Equity is completely at risk should the venture fail. Higher risk exposure and different income tax implications than loans make equity more expensive than debt to attract. Equity thus raises the WACC and hence the project cost, but is needed to establish project credibility, especially if the sponsors have poor records at cost control or low credit ratings, or the plant is the first of a kind or first in a country. Hence utilities are usually expected to channel significant equity into nuclear power plant investments.

Debt is borrowed money. Creditors are attracted by the creditworthiness of the project (potential for repayment) and the price (the cost of the loan and the risk–return ratio of the interest income offered to the creditor). The price or interest rate is commensurate with the perceived risk of the loan as well as with the presence of, and potential recourse to, collateral assets of the utility. If a creditworthy government or other entity guarantees the debt, the risk of non-payment and hence the cost of debt both fall significantly. Creditors by law have priority over owners in case of project failure. The fact that most loans involve contractually agreed interest rates and repayment schedules which are independent of plant performance further reduces the risks to lenders (NEA, 2009).

Proper conditions and incentives for attracting these elements would include assurances that the project is viable. This means that revenues will cover costs (which presumes careful market analysis); that profitable return of and on investment is assured (i.e., no cost overruns that reduce returns over the life of the project, and a regulatory and fiscal climate that is reasonably stable and not expropriative); that profits can be repatriated, if this is applicable; that debt repayment is guaranteed; and that risks are properly allocated and managed. Any viable financing scheme must include an effi-

[10] This section draws heavily on *Financing New Nuclear Power Plants* (IAEA, 2007).

cient and proper allocation of costs, risks, rights and responsibilities among the responsible parties. A project structure that imposes serious discipline in cost and risk management is a *sine qua non* of successful financing, whatever arrangements are made with regard to debt and equity.

For any sizable project, some combination of debt and equity is generally required; for multi-billion dollar projects like nuclear power plants, 100% equity or internal financing is highly unlikely. Debt will be preferred by project sponsors: the financing costs of attracting debt are lower than the costs of attracting equity, and debt puts someone else's money at risk. Lenders to the project will prefer a high equity component, to reduce their own exposure, and as a measure of credibility and project sponsor confidence or good faith. The split between equity and debt in the structure of any financing scheme will depend *inter alia* on the nature and financial position of the project sponsors, on local conditions where the plant is to be built, and on the viability, structure and evolution of the electricity sector in which the plant will operate. Many financing considerations are the same regardless of whether a plant's sponsors are state-owned companies or governments or private sector companies. However, the risks can be quite different.

Financing a nuclear power plant requires the commitment of large amounts of capital over extended periods of time. Payback periods of 30 years or more are substantially longer than those of most other generation technologies. Expenditures commence up to 10 years and more before the first revenue is obtained. Only a few private sector utilities or large institutional investors are willing and able to deal with the inherent uncertainties associated with such long payback periods (demand, market structures, prices, regulatory changes or policy interventions).

Financing nuclear power in the past

Unsurprisingly, governments have often taken the lead in promoting, developing and financing nuclear power. Nearly all nuclear power plants operating today were financed and built in regulated utility markets. In fact, much of the financing was provided by governments or with government backing or government guarantees of some kind. They have also used regulatory power to permit utilities building new plants to partially finance construction through the electricity tariff during the construction period ('allowance for funds used during construction', AFDC).

There were also cases where private finance coexisted with government financing. For example, in the USA and Germany, commercial financing has been arranged by private sector sponsors. Some plants, for example in France and the UK, were built by government-owned national utility companies, some of whose shares are publicly traded. And in some countries, like the Republic of Korea, nuclear plant financing has evolved over time from fully government financed to financing that, despite government ownership of Korea Hydro and Nuclear Power, is subject to commercial rules and conditions so as not to distort an otherwise liberalized market.

For private sector entities engaged in nuclear power plants, government involvement and regulated markets guaranteed a firm customer base and electricity prices sufficiently high to assure a profitable return. Under these conditions, cost overruns and project delays were covered by higher electricity prices and ultimately paid for by customers or from government budgets, thus minimizing the economic risk exposure of investors.

Financing today

In the last three decades both the utility and financial markets have changed in important ways. On the utility side, the rules have changed substantially. The new conventional wisdom is that progress means deregulating quasi-monopolistic markets and unbundling transmission, distribution and generation so that there is full competition among electricity generators and full choice for customers. While full deregulation, unbundling and competition are not yet established in most countries, this model affects financing considerations for new power plants. Thus the market risk for utilities has changed and will continue to change, even as demand for their product, electricity, continues to grow. Moreover, in liberalized energy markets, investment has become a private sector affair, again with direct implications for finance.

On the financial side, international capital markets have become increasingly global and competitive. While the basic types of equity and debt finance have not changed, a variety of new financial instruments and packaging schemes have evolved to better mitigate risk exposure, assure returns on investments and attract investors to specific projects. Access to global capital markets can be beneficial for public and private sector utilities alike, though it also has its downsides of being subjected to its short-term whims and market conditions.

An investment in a nuclear power plant will normally be led by a large utility or special-purpose entity. Other utilities, large-scale electricity users, vendors or simple investors may join the venture for different motives. Other utilities may wish to expand their portfolio by taken ownership and selling their share of electricity or gaining experience with nuclear power. Large-scale electricity users may wish to secure long-term (and low-carbon) supplies at predictable costs. Vendors may participate as part of the sales package, thus not only easing finance for the utility but also assuming a role in risk sharing. Straightforward investors provide finance with the objective of earning adequate returns.

Options

There are three basic ways in which a plant construction project can be structured, all of which have been used in the power sector: government (sovereign), corporate (balance sheet) and limited recourse (including project) finance (shielding sponsors' non-project assets from liability for project obligations). Government finance can come either straight from the annual budget, from government-issued securities (e.g., bonds) or from funds borrowed by the government in national or international capital markets. The terms of any non-budget approach depend on the country's overall credit rating. The government-owned utility will be the owner (and likely operator) of the plant. Any future operating profit will go to the government budget. Direct government involvement in a nuclear power project, e.g., asset ownership, equity participation, risk sharing and provision of various incentives including loan guarantees, imposes a certain degree of risk on the public sector itself (and thus society at large). The government may also incur indirect or non-finance-related risks, such as obligations to maintain infrastructures or assume the liability for plant and site decommissioning and spent fuel waste management.

Corporate financing means financing the project from the utility's (and partners') own resources, i.e. accumulated undistributed past profits, current revenue and from loans taken against existing assets. All participants (except lenders) will directly own and share the plant as an asset. The lead utility is the likely plant operator but will have to share the net proceeds (or whatever the arrangements foresee) with its partners. From the perspective of lenders, balance sheet finance secures their loans against all the assets of the utility and partners, not just the investment project. Plant owners may be able to exclude a portion of their assets from serving as potential collateral by ring-fencing parts of their corporate structures. But limiting the collateral increases the lenders' risks and lenders, in turn, will demand higher returns (interest) or decline providing loans at all.

The utility (and co-owners) assume the bulk of the investment risk against their asset base. Any problems with the plant such as construction delays, plant completion, commissioning or operational availability places these assets directly at risk. The financial sector tends to respond to a utility's nuclear investment decision by downgrading its credit rating, increasing its cost of borrowing across the board. With a likely (nominal) investment outlay of $10 billion to $14 billion for a twin-unit nuclear power station, a complete failure would put most utilities at the brink of bankruptcy.

Limited or non-recourse finance (also known as project finance) involves the foundation of a separate corporate entity for the sole purpose of constructing a power plant to be either sold after completion or operated for future revenue generation. Participation in the project occurs by putting up equity (i.e., buying shares) in the corporation. The corporation may seek loans for the plant construction from the financial sector or private investors but, given that the collateral is limited to the shares in the corporation itself (in other words the plant), the prospects for loans are generally slim or exquisitely expensive. Shareholders in the corporation, however, only risk the equity they put into the project while their other assets are protected.

Project sponsors do have some options for generating equity among themselves, either as good-faith money or to supplement available investment. One source of equity could be balance sheet financing. Another possibility could be to expand the number of equity partners to include partners who could provide equity in kind, or for principal customers to become major shareholders as a way of assuring security of supply. For Olkiluoto-3 in Finland, this latter approach made possible a 25% equity share. Another mitigating option is for sponsors to recruit local equity financing for local content.

The key differences among them are the ownership pattern they establish, which in turn governs the degree to which they protect the interest of investors and creditors, and the ways in which they allocate risk. Theoretically, any combination of entities, financing schemes and debt and equity could be considered for investment in the electricity industry, or for a nuclear power plant. In practice, this has not been the case. Non-recourse or limited-recourse financing, for example, offers no recourse collateral to lenders except the future income and assets of the project itself, and so tends to be used for renewable energy or less capital-intensive projects with shorter construction times and more flexible assets (e.g. natural gas turbines), rather than for capital-intensive investments like hydro projects and nuclear power plants. Schemes like public–private partnerships (PPP), build–operate–transfer (BOT), build–own–operate (BOO), and their variations, define the ultimate ownership of a project but are not really financing schemes (other than transferring finance obligations from the government-held utility to the private sector entity or partner in the investment venture).

Partial government involvement

Governments in developed and developing countries alike have increasingly found available budgets insufficient to meet all competing demands, and increasingly must turn to capital markets for financing specific projects or programmes; construction of new nuclear power plants would most likely fall into this category.

Even if a government does not build and own a new nuclear power plant, it can still take an equity share. If national budget resources are unavailable for this purpose, a government can create and dedicate government equity.

There are many ways in which a government can create equity. It can, for example, pledge receivables from creditable government-owned industries (or from industrial customers in the case of a government-owned utility); dedicate a portion of a government revenue stream (e.g. from mineral exports or taxes); pledge an asset like uranium reserves; barter (e.g. trade financing for agricultural exports); or pledge a service (like waste management). To the extent that a government uses this equity for, or otherwise assists in the financing of, a nuclear power plant, this might be considered as a subsidy or an unfair advantage for nuclear power under competition or trade rules in some jurisdictions. Other types of incentives or penalties to achieve desirable results, for example through contracting, might be structured to avoid this complication. However, to the extent that government participation involves government procurement, project costs will escalate: one World Bank estimate suggests that public procurement can add up to 40% to the cost of a project.

Other examples of possible government funding mechanisms include earmarked surcharges on all electricity sales, use of the national funds (for example, infrastructure funds or postal savings), creation of a government-run private bank to help finance 'clean energy projects' (including nuclear), banks to finance infrastructure, asset pooling (in countries or by utilities with other significant power generation assets), and (in developing countries) use of remittances from expatriates. Regional approaches, involving more than one government or utility, may also be used for financing nuclear power plants. Clearly, innovation and government financing are not mutually exclusive, nor are government and commercial financing.

Initial financing arrangements for a new nuclear plant might include some government funding for energy assessments and pre-construction studies or nuclear regulatory and legal infrastructures as well as research and human resource development; and capital market issues of financial instruments (securities, stocks, bonds). For plants in developing countries, additional resources could include directly allocated development funds from international aid organizations and development banks, or other government-sponsored aid programmes, Export Credit Agency (ECA) insurance schemes or institutions like the Overseas Private Investment Corporation (OPIC) and the Multilateral Investment Guarantee Agency (MIGA) (although these only ensure that the suppliers of the equipment but not the project sponsors get paid in case of delays or default), and equity investments and commercial loans. Many within the nuclear community assert that multilateral banks should become directly involved in financing nuclear plants. However, multilateral banks are required to balance the views of their Member States, which have strong and diverse views on nuclear power. Moreover, as banks, their investment criteria include demonstrating that a proposed nuclear plant will be the least-cost alternative for electricity generating capacity expansion, and/or cost efficient for solving environmental, security and other social problems, if these are included in a government's project proposal.

Government support for debt has consisted primarily and traditionally of providing loan or other types of guarantees to facilitate financing of large infrastructure projects. If structured to the benefit of the government as well as the recipient, loan guarantees can be a source of revenue rather than a subsidy/cost to the government. Using an insurance scheme or export credit approach, governments could, for example, charge interest on the size of the loan as the price of the guarantee. Guarantees can also include guaranteed power purchases (take-or-pay contracts), or even agreements to cover costs of delay arising from government action or inaction. Each of these guarantees carries its own risks for the government, which then becomes liable for non-performance, perhaps as the result of something over which it has no control. Governments in Asia readily entered into highly optimistic purchase power agreements to secure project financing for needed power plants, only to find that slower economic growth after the Asian economic crisis of 1997 made fulfilment of these obligations impossible. Some Latin American countries in the 1980s secured loans in hard currency for projects whose revenues were in local currency, only to have exchange rates shift dramatically, forcing default on large loans. Such guarantees are not unique to government – they can also be, and variously have been, provided by utilities, other large corporations or consortia of companies. The risks would be the same, but the losses would accrue to private investors and not to the government.

15.3 Levelized cost of electricity generation

Numerous studies routinely assess the current and future competitiveness of different electricity generating options under different scenario assumptions. In a wide range of scenarios, nuclear power is a least-cost option for centralized base-load electricity generation (ENEF, 2010; NEA and IEA, 2010). The economic performance of nuclear power versus its alternatives is highly dependent on numerous factors such as the costs and availability of natural gas and coal, hydro power resources or wind availability, which allow direct comparisons only on a clearly defined case-by-case basis. Some studies question the economic competitiveness of nuclear energy usually by generalizing worst practices and denying future learning to nuclear power while assuming best practices and rapid future learning to non-nuclear alternatives, especially renewables (EREC and Greenpeace, 2010; WISE, 2009a, 2009b; Schneider *et al.*, 2009).

In essence, because of the sometimes drastically divergent assumptions about the future driving forces of electricity demand and supply, technology and policy, the generating costs reported by these studies are unsuitable for comparisons. One exception is the already mentioned OECD report *Projected Costs of Generating Electricity* (NEA and IEA, 2010).

The OECD study calculates 'levelized cost of electricity' (LCOE) using two real discount interest rates, 5% and 10%,[11] applied to all technologies, harmonized generic technology performance assumptions and boundaries, and clearly specified fuel prices. For the first time, the study assessed the impact of a carbon price of $30 per tonne of carbon dioxide. The generating cost calculations, based on the simple levelized average (unit) lifetime cost approach based on the discounted cash flow (DCF) method, are summarized in Fig. 15.11.

The study reached two important conclusions. First, at low discount rates, capital-intensive generating technologies such as nuclear energy are among the least-cost baseload generating options. The actual merit order is location dependent and cannot be generalized.

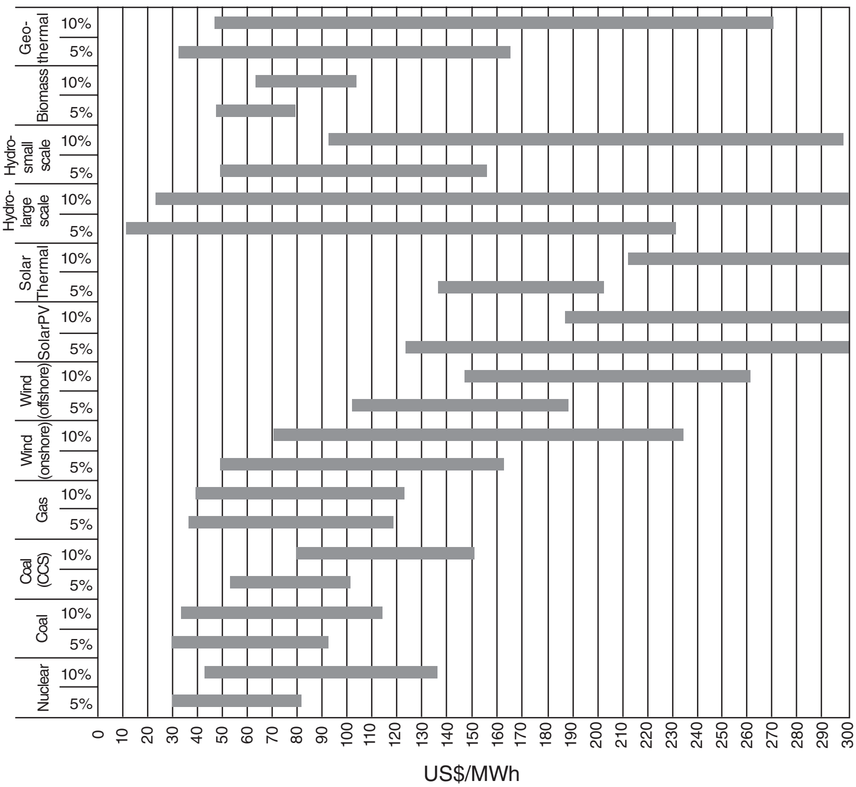
An exception is provided by locations with lowest-cost coal availability, e.g. Australia or certain parts of the USA or (although not part of the OECD study by analogy) parts of China, India and other coal-rich developing countries. Here coal, even when equipped with carbon capture, outperforms nuclear power. A similar observation is valid for hydro power.

Second, at 10% discount rates, the competitiveness of nuclear power slips and fossil generation gains on nuclear power. In some locations, coal with and without carbon abatement as well as CCGT are least-cost generators. In others nuclear maintains its overall cost-competiveness.

The calculations highlight the paramount importance of discount rates, and to a lesser extent carbon and fuel prices when comparing different technologies (NEA and IEA, 2010).[12]

[11] A key limitation of the LCOE is that it does not take into account the different levels of risks among investment alternatives (NEA and IEA, 2010). Interest rates demanded by investors and lenders reflect the opportunity costs of money as well as the perceived risk of an investment. Put differently, private sector capital gravitates to projects that offer highest returns commensurate with associated risks. And investment in electricity generation competes with (non-electric) alternatives in the global capital market. The commensurate risks of a particular investment are better captured by applying WACC – which accounts for the split between equity and debt financing and associated costs. To some degree, technology-specific risks are captured by the split with higher risk projects demanding higher equity shares and thus higher cost financing. It has been argued that the high capital risks associated with new nuclear construction may lead to higher cost of debt than other conventional power plant projects (KPMG, 2010).

[12] The generating cost range of Fig. 15.6 excludes any costs or taxes on carbon emissions.

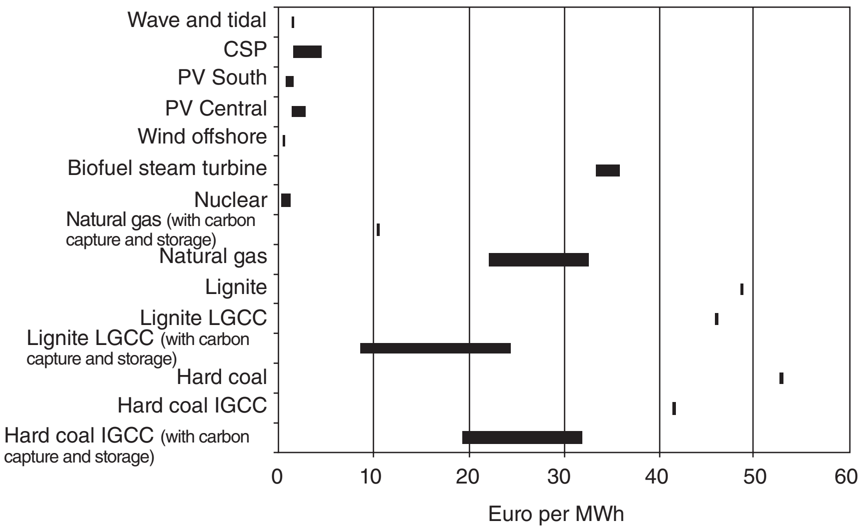


15.11 Expected generating cost of different generating options (without carbon dioxide taxes): CSP = concentrating solar power, PV = photovoltaic, CCS = carbon capture and storage, IGCC = integrated gasification combined cycle. Adapted from NEA and IEA (2010).

15.3.1 Externalities

While currently not included in standard electricity cost accounting schemes, decision makers should be aware of cost factors imposed on the public by the production and use of electricity. These costs are real and a fair share have directly and indirectly been compensated by the public purse (or resulted in reduced government revenue). Since investors normally do not consider externalities in investment decision making, it falls upon government policy to 'internalize the external costs' of the health and environmental damages resulting from power generation. In fact, in the past, internalization has been imposed on electricity generation[13] but insuffi-

[13] Examples include the regulation of emissions of some pollutants from fossil fuel combustion (EU Directive 2001/80/EC), the Clean Air Act and its various amendments in the US, the sulphur emission limitations ruled by the State Environmental



15.12 External costs of different generating options. Adapted from Preiss and Friedrich (2009).

ciently by far for a full internalization.[14] The most recent studies addressing life-cycle externalities from electricity generation show nuclear power as one of the technologies with the lowest externalities (Preiss and Friedrich, 2009; NRC, 2009). One of the externalities of nuclear power is the cost of a severe nuclear accident (e.g., Chernobyl or Fukushima). These are calculated on a probabilistic basis (low probability – high consequence) and given the large amount of kWhs produced by nuclear power plants are still small despite the enormous damage costs of an accident. Figure 15.12 summarizes the findings of the NEEDS study (Preiss and Friedrich, 2009). Clearly, factoring these externalities into the price of electricity would fundamentally change the merit order of generating options in favour of nuclear power and renewables.

Protection Administration (SEPA) in China, Directive 2001/77/EC of the European Parliament and of the Council of 27 September 2001 on the Promotion of Electricity Produced from Renewable Energy Sources that mandates the integration of more expensive, non-dispatchable electricity from renewables, the European Emissions Trading Directive (2003/87/EC), the Price-Anderson Indemnity Act which governs liability-related issues for all non-military nuclear facilities in the United States, the EU Directive on Nuclear Safety (2009/71/EURATOM) and the Kyoto Protocol limiting greenhouse gas emissions in industrialized countries.

[14] There is a host of issues yet to be resolved ranging from attribution of damages to their monetary valuation.

15.4 Risks and uncertainties

From the foregoing, one can derive five principal risk areas for utilities and sponsors of nuclear power projects, i.e., planning, construction, electricity market rates, operational, and waste management/decommissioning (CITI, 2009).

15.4.1 Planning

The lead times from the decision to build a new nuclear power plant until breaking ground are usually measured in multiple years up to a decade. Site selection, acquisition and regulatory site approval, obtaining bids from vendors and bid evaluation, stakeholder involvement and finance arrangements are time-consuming steps and often have to be carried out sequentially rather than in parallel. Recent early site permits in the US took three to four years from application until the permit was granted. Moreover, nuclear power remains controversial in many jurisdictions and opposition to new developments often results in lengthy hearings and court involvements and thus extended planning timelines. Many governments have recognized the added uncertainties in lengthy planning processes and have begun to revise and streamline procedures so as to expedite lead times and reduce uncertainties for nuclear power project sponsors. From a financial perspective, the planning uncertainty faced by developers is the least risky element and no real threat to the financial integrity (CITI, 2009). Still, a utility might have spent several tens of millions on site acquisition, design certification and legal procedures. In addition, a denied site or construction certification after several years may put the utility in a position short of generating capacity with the need to resort to costly alternative suppliers.

15.4.2 Construction

Plant construction completion on time and on budget is by far the largest financial risk faced by investors in nuclear power. With a financial exposure of several billion dollars per plant, even small cost overruns or slippages in completion can adversely affect a utility's equity value. The negative examples of the cost overruns of an estimated 75% and completion delay of three to four years as experienced by the Olkiluoto project in Finland (NW, 2010a; KPMG, 2010) and the 50% over budget and two years behind schedule of the Flamanville-3 plant in France (NW, 2010b) led many financial analysts to conclude that the 'economics of nuclear say no' to new nuclear build (CITI, 2009; Moody's, 2009). Clearly, such cost overruns can only be shouldered by the largest utilities such as TVO or EDF. Both entities are special in their ownership structures, which differ greatly from other utilities around the world and have helped avoid otherwise likely economic and financial troubles. TVO, though a privately held company, sells its electricity exclusively to the owners at cost, which eliminates some uncertainties such as demand and market risks and also allows passing through unexpected higher generating costs. EDF is largely government owned (84.5%) and thus better bolstered for events like Flamanville than privately held utilities operating in competitive markets.

The track records of Olkiluoto and Flamanville are worrisome indeed, but one has to put them into the perspective of a FOAK situation and the lessons learned for future projects. Many potential nuclear power undertakings in Europe and North America are likely to encounter some kind of FOAK flavour simply because of lack of recent construction experience. This and the fact that, historically, many nuclear-building utilities suffered downgrades in their credit ratings during the construction phase (Moody's, 2009), reflecting the risk profile of nuclear power investments, are arguments used by finance institutions in their pessimistic outlook on new nuclear build.

One suggested hedge for containing construction cost overruns is phased financing. This approach, already implemented in China and proposed for new plants in the US, involves financing a project in tranches, starting with construction. The cost of capital for each phase will reflect the risks only of that phase, so that the high costs of construction risks are not carried over throughout the project. During construction, the main risk is completion on time and within budget. As construction proceeds and completion risks diminish, the cost of capital can also fall. Once completed, investor risks are essentially reduced to operational and market risks (revenue stream). Different financing phases may also have different capital structures: for example, shareholders would generally be at risk for the construction phase, but non-recourse financing might be introduced with the onset of commercial operations. Phased financing is deemed to be especially effective with a phased asset transfer and, where applicable, a phased sell-out of government interests. Phased financing may thus facilitate government participation in a private sector project, since a government could choose to finance or guarantee only a part of the project and then privatize its share of the plant. The concept of phasing may also help to manage supply bottlenecks and the need for trained personnel, regulators and other project inputs.

The same concept of phasing applies on a broader scale to the start of a nuclear programme. The first unit will carry a higher risk of successful completion – and higher costs – than subsequent units. However, once a few units are built and operating successfully, the financing model can change, with revenue from operating units being used to finance new build.

But it can confidently be expected that, with regained knowledge and build experience, construction schedules will be met and cost overruns minimized as demonstrated elsewhere: neither the limited European construction experience nor the nuclear power history of North America is representative globally. There are numerous nuclear power plants in Asia that have been completed on time and on budget. One can only move down the learning curve with repeated plant construction over short time intervals. Construction times of just 48 months or four years have been demonstrated in the Republic of Korea, Japan and China – these three countries alone account for more than 70% of all nuclear power construction activity since 1990.

However, in most industrialized countries new construction of power-generating plants has generally been limited and has lacked technological diversity. In the last decade, the majority of new generating plant constructed in non-Asian OECD countries has been either gas (especially combined cycle gas turbines) or new renewables, especially onshore wind (NEA and IEA, 2010). So new coal power plants, especially if equipped with CCS, share the issue of construction cost uncertainty with nuclear power.

15.4.3 Market rates

Irrespective of the actual market structure – liberalized or regulated – the cash flow and profitability of a utility depend on its operating efficiency and the price at which it can sell its electricity in the marketplace. The high fixed costs and low operating costs of new nuclear power plants require higher revenue per kWh to break even than most competing alternatives. It is questionable if private sector entities involved in nuclear power projects are willing to take on the price risk. In regulated markets of developing countries, social considerations of delivering affordable electricity to the poor are often enforced upon utilities to sell electricity below costs. Their economic survival then hinges upon government subsidies. In either situation, the price risk serves as a barrier for private sector finance of new nuclear build.

Market risks can be mitigated with long-term power purchase agreements with large-scale electricity customers such as electricity-intensive industries and larger communities. It has been argued that with long-term power purchase agreements in place, lending institutions would be satisfied with an expected rate of return of 5% to 10%. The same plant operating in a merchant market with no underpinning contracts would be confronted with rates of 10–12% (Bulleid, 2005) with direct implications for WACC and IDC.

Generally, once operating and with plant completion risks eliminated, the economics of nuclear power plants are viewed favourably by ratings agencies and investors alike. The longer-term outlook is even better, when plants are more and more amortized and the capital portion of operating costs approaches zero.

Environmental policy is another uncertain element influencing the market price of electricity. Nuclear power has a small greenhouse gas (GHG) footprint per kWh, thus its competitiveness (along with other low GHG-emitting technologies) would benefit from policies targeted at mitigating climate change. Electricity demand prospects themselves are a source of uncertainty. The emergence and market penetration of smart grids, including real-time pricing, may flatten the load profile – a positive aspect for the baseload technology nuclear power – but also better integrate intermittently available renewables, thus improving their competitiveness against nuclear power. Efficiency improvements at the level of electricity use spurred by government policy could substantially dampen future demand growth while a large-scale advent of electric vehicles might even result in accelerated growth.

15.4.4 Operational

Operational risks relate primarily to operational unreliability due to unplanned outage. High fixed costs combined with unit sizes often counted in multiples of fossil and renewable plant capacities make the unavailability of a nuclear generating station a costly affair. In addition to lost revenues, utilities that sold their electricity under long-term power purchase agreements may be forced to provide high-cost replacement power from other generators. Operational risks are generally less an issue for utilities with a sufficiently large portfolio of generating capacities.

Plant operating safety is a non-negotiable prerequisite for a profitable nuclear power plant. A plant that is found to be not in compliance with operating safety regulations will be shutdown by the national regulatory authority and a shutdown plant does not earn revenues. Moreover, regulatory oversight and, if necessary, intervention also protects the utility's revenue generating asset from potential serious damage and long-term unavailability. An operational risk exists, however, if regulatory intervention is politically motivated and not exclusively safety related.

15.4.5 Waste and decommissioning

Private sector investors shy away from unknown or unknowable liabilities. Spent fuel and nuclear waste management, as well as plant decommissioning at the end of a service life of 60 or more years, are factors with no practical or commercial evidence (except for decommissioning) regarding their eventual costs. It is also unknown under what kind of regulatory environment waste management and decommissioning will take place, e.g., to what level will plant sites have to be decommissioned beyond plant demolition, decontamination and debris removal. In order to cope with long-term liabilities, most jurisdictions assess a levy on nuclear power plant operators for every kWh produced to be paid into an escrow fund (or equivalent) to be used for waste management and decommissioning. Whether or not the escrow funds accumulate funds sufficiently large to cover all post-closure cost remains to be seen but their existence limits the risk exposure of investors.

15.5 Conclusions

Generally, the economic prospects of nuclear power look promising, and generating costs on a life-cycle basis are competitive against alternatives in many markets. But nuclear power is capital intensive with long amortization periods and capital requirements that amount to several billion dollars per unit – overstretching the comfort levels of many investors. Finance, therefore, is one of the major barriers for nuclear power. In liberalized markets only very large utilities can finance a nuclear power project.

The economics of nuclear power embrace more than the life-cycle generating costs and include energy supply security, reliability and price stability considerations as well as environmental policy objectives. Nuclear power is a technology with the lowest externalities – as most externalities have already been internalized. Nuclear power is an effective and efficient GHG mitigation technology. Where energy security and protection of the environment are national policy objectives, a quasi-internalization of externalities may warrant some form of financial support or guarantee for private sector investment in new nuclear plants. A level playing field with clear and uniform performance criteria for all generating options reduces overall uncertainty and raises the probability that electricity market prices over the plant's lifetime will provide an adequate return on investment.

At the minimum, unambiguous and sustained government policy support is required for nuclear power to unfold its economic potential. Such a policy in support of national nuclear power programmes as an integral part of a national energy strategy is paramount for investor and lender confidence and public acceptance of the technology.

Future international GHG reduction schemes may also recognize the mitigation potential of nuclear power and thus increase its attractiveness to investors and lenders, particularly schemes that award emission credits for environmentally benign investments abroad.[15] But even here, economic

[15] Finance of nuclear power plant under the flexible mechanism under the Kyoto Protocol, the Clean Development Mechanism (CDM) and Joint implementation (JI) for the purpose of acquiring GHG emission credits is presently disallowed. If governments wish to utilize such finance mechanisms, this needs to be reflected in the post-2012 international environmental agreement currently being negotiated under the aegis of the United Nations Framework Convention on Climate Change (UNFCCC).

viability is inescapable; no-one is likely to invest in a financial black hole, nor to build nuclear power plants for environmental reasons, unless they are demonstrably profitable and among the most cost-efficient solutions.

The global financial community is still attributing a deterring risk/reward ratio to nuclear power. International organizations and governments alike need to join hands in enhancing the community's ability to assess the investment risks involved in nuclear power projects so that it can provide suitable finance packages for such investments, especially for countries currently without active nuclear power programmes. Newcomer countries will depend on the assistance of technology holders in launching their national nuclear programmes. Nuclear infrastructure and human resource development followed by financing are key in this regard.

The capital costs of nuclear power are expected to further improve as the number of plant orders increase and FOAK conditions decrease. The cost reduction potential for technology learning but also for design standardization is substantial.

Planning and construction times of nuclear power plants are longer than for most alternatives, excluding nuclear power from quick-fix solutions. Nuclear power is not a quick-fix solution to a country's energy problems. But as an integral part of a long-term energy strategy, nuclear energy can contribute to a country's sustainable energy development objectives.

15.6 References

Bohi DR and Toman MA (1996), *The Economics of Energy Security.* Dordrecht, The Netherlands, Kluwer Academic Publishers.

Bulleid R (2005), Nuclear power – a financial reaction, *Environmental Finance* 6(9): 20–21.

Bunn M, Fetter S, Holdren JP and van der Zwaan B (2003), *The Economics of Reprocessing vs Direct Disposal of Spent Nuclear Fuel*, President and Fellows of Harvard University, Cambridge, MA.

Canterbery E, Johnson B and Reading D (1996), Cost savings from nuclear regulatory reform: an econometric model, *Southern Economic Journal* 62: 554–556.

Cantor R and Hewlett J (1988), The economics of nuclear power: further evidence on learning, economies of scale, and regulatory effects, *Resources and Energy* 10: 315–335.

CBO (2008), *Nuclear Power's Role in Generating Electricity*, Congressional Budget Office, Washington, DC.

CITI (2009), *New Nuclear – The Economics Say No*. CITI Investment Research & Analysis, New York.

Copulos MR (2007), The hidden cost of imported oil – an update. The National Defense Council Foundation, downloaded 11 May, 2009 from www.ndcf.org.

Deutch J and Moniz EJ (2003), *The Future of Nuclear Power: An Interdisciplinary MIT Study*. Massachusetts Institute of Technology, Cambridge, MA, http://web.mit.edu/nuclearpower.

Deutch J, Forsberg CW, Kadak AC, Kazimi MS, Moniz EJ and Parsons JE (2009), Update of the MIT 2003 Future of Nuclear Power Study – An Interdisciplinary

MIT Study, Massachusetts Institute of Technology, Cambridge, MA, http://web.mit.edu/nuclearpower.

Du Y and Parsons JE (2009), *Update on the Cost of Nuclear Power*, MIT Center for Energy and Environmental Policy Research (CEEPR), Cambridge, MA.

ENEF (2010), Strengths–Weaknesses–Opportunities–Threats (SWOT) Analysis, ENEF Working Group Opportunities – Subgroup on Competitiveness of Nuclear Power, European Nuclear Energy Forum of the European Commission, Brussels.

EREC and Greenpeace (2010), Energy [r]evolution: A Sustainable World Energy Outlook, 3rd edition, 2010 World Energy Scenario. European Renewable Energy Council (EREC), Brussels.

Hosie J (2007), *Turnkey contracting under the FIDIC Silver Book: What do owners want? What do they get?* Mayer Brown International LLP, London.

IAEA (2007), *Financing New Nuclear Power Plants*, IAEA Nuclear Energy Series no. NG-T-4.2, International Atomic Energy Agency, Vienna.

IAEA (2010), *International Status and Prospects of Nuclear Power*, GOV/INF/2010/12-GC(54)/INF/5, International Atomic Energy Agency, Vienna.

IAEA (2011), *Power Reactor Information System (PRIS)*, International Atomic Energy Agency, http://www.iaea.org/programmes/a2/index.html, accessed 30 November 2011.

IPCC (2007), *Climate Change 2007: Mitigation*. Contribution of Working Group III to the Fourth Assessment Report of the Intergovernmental Panel on Climate Change (Metz B, Davidson OR, Bosch PR, Dave R and Meyer LA, eds), Cambridge, UK, Cambridge University Press.

Komanoff C (1981), *Power Plant Cost Escalation*. Komanoff Energy Associates, New York.

Kozlov VV (2004), Determination of the highest possible capital costs for construction of a nuclear power plant based on data from construction abroad, *Atomic Energy* 97(5): 338–345.

KPMG (2010), *Investment in Nuclear in the Context of Low–Carbon Generation*. KPMG LLP, London.

McCabe MJ (1996), Principals, agents, and the learning curve: the case of steam-electric power plant design and construction, *Journal of Industrial Economics* 44: 357–375.

Moody's (2009), *New Nuclear Generation: Ratings Pressure Increasing, Special Comment*. Moody's Investor Services. New York.

Mooz WE (1979), *A Second Cost Analysis of Light Water Reactor Power Plants*. The Rand Corporation, R-2504-RC, Santa Monica, CA.

NEA (2003), *Nuclear Energy Today*. Nuclear Energy Agency of the Organization for Economic Co-operation and Development, Paris.

NEA (2009), *The Financing of Nuclear Power Plants*. NEA no. 6360, Nuclear Energy Agency of the Organization for Economic Co-operation and Development, Paris.

NEA (2010), *Uranium 2009: Resources, Production and Demand*. A Joint Report prepared by the OECD Nuclear Energy Agency and the International Atomic Energy Agency, Nuclear Energy Agency of the Organization for Economic Co-operation and Development, Paris.

NEA and IEA (2005), *Projected Costs of Generating Electricity – 2005 Update*. Nuclear Energy Agency and International Energy Agency of the Organization for Economic Co-operation and Development, Paris.

NEA and IEA (2010), *Projected Costs of Generating Electricity – 2010 Update*. Nuclear Energy Agency and International Energy Agency of the Organization for Economic Co-operation and Development, Paris.

NRC (2009), *Hidden Costs of Energy: Unpriced Consequences of Energy Production and Use*. Committee on Health, Environmental, and Other External Costs and Benefits of Energy Production and Consumption; National Research Council, National Academy of Sciences, Washington, DC, http://books.nap.edu/catalog/12794.html.

NW (2010a), Olkiluoto-3 operation again delayed, but progress made on I&C issue, *Nucleonics Week,* Vol. 51, No. 23, 10 June 2010.

NW (2010b), Flamanville-3 two years behind schedule, EDF says in targeting 2014 for startup, *Nucleonics Week,* Vol. 51, No. 31, 5 August 2010.

Paik S and Schriver WR (1979), The effect of increased regulation on capital costs and manual labor requirements of nuclear power plants, *The Engineering Economist* 26: 223–244.

Preiss P and Friedrich R (2009), *Report on the application of the tools for innovative energy technologies*, New Energy Externalities Developments for Sustainability (NEEDS), Technical Paper no. 7.2 – RS 1b project no: 502687, http://www.needs-project.org/2009/TechnicalPapers/NEEDS_Rs1b_WP7_TP7_2_final.pdf.

Reuters (2010), Progress ups Levy nuclear plant costs, delays start, Reuters 6 May 2010. http://www.reuters.com/article/2010/05/06/utilities-progress-levy-idUSN0611303620100506

Rogner H-H (2007), *Uranium*, Survey of Energy Resources, World Energy Council 2007, London.

Rogner H-H (2010), *Uranium*, Survey of Energy Resources, World Energy Council 2010, London.

Roth IF and Ambs LL (2004), Incorporating externalities into a full cost approach to electric power generation life-cycle costing, *Energy* 29(12–15): 2125–2144.

Rothwell G (2010), *New U.S. Nuclear Generation: 2010–2030*. Backgrounder. Resources for the Future, Washington, DC.

RWE Nukem (2002), *RWE Nukem Market Report 2002*. RWE Nukem Inc., Danbury, CT.

Schneider M, Thomas S, Frogatt A and Koplow D (2009), 2009 world nuclear industry status report, *Bulletin of the Atomic Scientists* 65(6): 1–19.

Scoggs S (2007), Direct Testimony. In Re: Florida Power & Light Company's Petition to Determine Need for Turkey Point Nuclear Units 6 and 7 Electrical Power Plan, Florida Public Service Commission, 16 October.

Tyner WE (2007), Biofuels, energy security, and global warming policy interactions, paper presented at the National Agricultural Biotechnology Council Conference, South Dakota State University, Brookings, SD, 22–24 May 2007.

UN (1992), *United Nations Framework Convention on Climate Change*. United Nations, New York.

UN (1998), *Kyoto Protocol to the United Nations Framework Convention on Climate Change*. United Nations, New York.

UoC (2004), *The Economic Future of Nuclear Power – A Study Conducted at the University of Chicago*. http://www.ne.doe.gov/np2010/reports/NuclIndustryStudy-Summary.pdf.

Weisser D (2007), A guide to life-cycle greenhouse gas (GHG) emissions from electric supply technologies, *Energy* 32: 1543–1559.

WISE (2009a), The economics of nuclear reactors: renaissance or relapse? *World Nuclear Monitor*, no. 692–693, World Information Service on Energy (WISE), Amsterdam, Netherlands, 28 August 2009.

WISE (2009b), The past as prologue: the persistent upward spiral of nuclear reactor costs, *World Nuclear Monitor*, no. 692–693, World Information Service on Energy (WISE), Amsterdam, Netherlands, 28 August 2009.

WNN (2010), Construction of Fangchenggang plant starts, *World Nuclear News*, 3 August 2010.

Zimmerman MB (1982), Learning effects and the commercialization of new energy technologies: The case of nuclear power, *Bell Journal of Economics* 13: 297–310.

16
Social impacts and public perception of nuclear power

F. BAZILE, CEA, France

Abstract: Social impacts of nuclear power are significant but difficult to quantify, as there is no consensus on a method. The first part of this chapter presents a review of the advantages and drawbacks of nuclear power compared to other power generation sources, as they have been assessed in recent publications. The second part presents public perceptions of nuclear power and tries to identify levers for a better acceptance. Beyond specific national issues, two main points can be identified. First, there is a link between education level, knowledge of energy matters and acceptance of nuclear power; in particular, knowledge of its potential contribution to a low-carbon energy mix, and an awareness of the physical limits of renewable energies (such as solar and wind) contribute to an acceptance of nuclear power. Second, the more concrete a knowledge of nuclear power people have, for example by living in the vicinity of a nuclear plant, the more they accept it, as its economic benefits and safe operation are better understood.

Key words: energy policy, economics, public perception of risk, safety, externalities, low-carbon energy mix, Chernobyl, radioactive waste management, opinion surveys, stakeholder involvement, public debate, political decision.

16.1 Introduction

The social impacts of choosing nuclear power have to be assessed from a long-term perspective, i.e. by a minimum of a century, or much more if one takes into account waste management. Fifteen years are needed between the decision to launch a nuclear build and the beginning of operation, with those 15 years including time to undertake all the political work to establish the infrastructure (such as the creation of a regulatory authority and the promulgation of an institutional and legal framework). The lifespan of operation can be about 50–60 years, perhaps even more with future designs, depending on the safety rules acknowledged in each country. Dismantling and decommissioning require several decades, depending on technologies and on the availability of waste management facilities. Whatever the length of time involved, choosing to include nuclear power in a country's energy mix is a political commitment and not just a technical decision.

Sustainable development is widely recognized, at an international level, as a relevant objective of energy policies. And it is agreed that three interrelated dimensions – economy, environment and social – need to be taken into account, and that there needs to be an equilibrium between present and future generations. Nuclear choice should be assessed from this perspective, since it has significant impacts on all these dimensions.

But the benefits resulting from choosing nuclear power are not always, or spontaneously, evident to the public at large. Indeed, nuclear energy often has a negative visibility, since many people perceive and overestimate the risk of major accidents (referring to Chernobyl or to Fukushima), the terrorist risk and uncertainties about waste management, and, moreover, because of the 'original sin' of nuclear technology – the nuclear bomb – and the risk of proliferation.

The first part of this chapter exposes some of the main issues regarding the social impacts of nuclear power, even if they are difficult to quantify and therefore possibly controversial. The second part focuses on public perception of nuclear power, including risk perception, as shown by opinion polls and qualitative surveys.

16.2 Social impacts at both national and local levels

Launching a nuclear programme has social impacts at different levels: at a national level it can mean a political choice regarding the energy mix and a carbon-free energy policy; at a local level, it can mean local development and employment on one hand, and environmental impacts on human health and nature on the other. Both these two levels need to be addressed. In a newcomer country, national public opinion needs to be prepared, which means providing educational information about the energy mix, and on the advantages and drawbacks of each energy source, and analysing nuclear power's risks and benefits from the perspective of a comparison with other sources of electricity generation, notably by distinguishing carbon-free sources (nuclear power, hydraulics and new renewable energies) and fossil sources. Such programmes need to give people objective information about all energy sources and not just about nuclear power. If nuclear power is considered without comparison to other sources, a large part of the public will probably focus on accident risks, on radioactivity's potential risk to human health, and on long-term radioactive waste – the main arguments developed by nuclear opponents everywhere in the world. All dimensions of energy policy need to be taken into account, including security of supply and the prevention of global climate change, and not just assessed over the short term.

Many studies have been implemented in order to help decision-makers plan an energy policy and to define the respective shares of different energy sources, particularly electricity generation sources. No approach benefits from a total consensus, and social impacts of the choice of energy source are the more controversial, since they are the most difficult to quantify. The tools proposed therefore have to be considered as an heuristic framework to discuss the different energy options, and to make the choices more transparent and open to debate. A comprehensive set of indicators to compare technologies is given by Hirschberg *et al.* (2004).

Table 16.1 provides a framework of indicators covering the main aspects of nuclear choice. The respective weight of each dimension is an important part of the political choice. They depend, of course, on the national context,

Table 16.1 Illustrative set of technology-specific indicators

<table>
<tr><td>Dimension</td><td colspan="2">Impact area</td><td colspan="2">Indicator</td><td>Unit</td></tr>
<tr><td>Economy</td><td colspan="2">Financial requirements</td><td colspan="2">Production cost</td><td>c/kWh</td></tr>
<tr><td colspan="6">Fuel price increase sensitivity</td></tr>
<tr><td colspan="2">Resources</td><td colspan="2">Availability (load factor)</td><td colspan="2">%</td></tr>
<tr><td colspan="3">Geo-political factors</td><td colspan="3">relative scale</td></tr>
<tr><td colspan="3">Long-term sustainability: Energetic resource lifetime</td><td colspan="3">years</td></tr>
<tr><td colspan="3">Long-term sustainability: Non-energetic resource consumption</td><td colspan="3">kg/GWh</td></tr>
<tr><td colspan="3">Peak load response</td><td colspan="3">relative scale</td></tr>
<tr><td>Environment</td><td colspan="2">Global warming</td><td colspan="2">CO_2-equivalents</td><td>tons/GWh</td></tr>
<tr><td colspan="2">Regional environmental impact</td><td colspan="2">Change in unprotected ecosystem area</td><td colspan="2">km^2/GWh</td></tr>
<tr><td colspan="2">Non-pollutant effects</td><td colspan="2">Land use</td><td colspan="2">m^2/GWh</td></tr>
<tr><td colspan="2">Severe accidents</td><td colspan="2">Fatalities</td><td colspan="2">fatalities/GWh</td></tr>
<tr><td colspan="2">Total waste</td><td colspan="2">Weight</td><td colspan="2">tonnes/GWh</td></tr>
<tr><td>Social</td><td colspan="2">Employment</td><td colspan="2">Technology-specific job opportunities</td><td>person-years/GWh</td></tr>
<tr><td colspan="2">Proliferation</td><td colspan="2">Potential</td><td colspan="2">relative scale</td></tr>
<tr><td colspan="2">Human health impacts (normal operation)</td><td colspan="2">Mortality (reduced life-expectancy)</td><td colspan="2">years of life lost/GWh</td></tr>
<tr><td colspan="2">Local disturbance</td><td colspan="2">Noise, visual amenity</td><td colspan="2">relative scale</td></tr>
<tr><td colspan="2">Critical waste confinement</td><td colspan="2">‘Necessary’ confinement time</td><td colspan="2">thousands of years</td></tr>
<tr><td colspan="2">Risk aversion</td><td colspan="2">Maximum credible number of fatalities per accident</td><td colspan="2">max fatalities/accident</td></tr>
</table>

Source: Hirschberg *et al.* (2004).

political stability, economic data, financing capacities, geographical constraints (primary resources, geopolitics, etc.), the country's development and growth, and so on. Countries like Japan or France, which have few or no fossil fuel resources, have a more evident need for nuclear power, for security of supply and to reduce costly imports of fossil fuels. However, the total costs of a nuclear programme must include what might be called 'infrastructure costs': human resources, a legal framework, a safety authority, perhaps an industrial supply chain, etc. In a non-nuclear country envisioning the launch of a nuclear programme, it is necessary to undertake an opportunity study, to assess energy and electricity needs and to compare the merits of each energy source. In 2007, IAEA published a guide for newcomers, known as *Milestones in the Development of a National Infrastructure for Nuclear Power*, which precisely exposes the infrastructure requirements needed, and indicates the steps needed to assess their readiness (IAEA, 2007).

16.2.1 Social impacts at a national level

At a national level, the main benefits of nuclear power are security of supply, steady costs of base-load electricity and a contribution to a low-carbon electric mix. It is difficult (and slightly artificial) to distinguish the social and economic impacts of nuclear power for a country. Access to electricity at a steady and low cost is an economic benefit, which results from the production costs of nuclear electricity, but there is also a social impact in access to electricity, particularly in developing countries, as it conditions development, health, access to knowledge, and so on. Building a nuclear plant is cost intensive but, in operation, the fuel cost represents less than 10% of production costs so that, even if the price of uranium increases, it will not have significant impact on the kWh production cost. According to a 2000 study in Finland, cited in WNA's *The Economics of Nuclear Power* (WNA, 2010), a doubling of fuel prices would result in the electricity costs from nuclear energy rising by about 9%, for coal by 31% and for gas by 66% (see also Chapter 15 on economics). For a newcomer country, in an opportunity study, it will be necessary to draw up forecasts of the country's energy demand (for instance, with high and low scenarios, taking into account growth of GDP), and to compare the competitiveness of the different electricity production options – whether coal, Combined Cycle Gas Turbine, or (possibly in oil countries) an oil-fired plant. In countries which produce high-value fossil sources (oil or gas), the revenues 'saved' by nuclear production and generated by exporting oil or gas also have to be taken into account. However, in the case of nuclear kWh production, all life-cycle environmental costs need to be internalized, notably including those associated with waste management and decommissioning (though if CO_2 emissions were priced, it would increase nuclear competitiveness against fossil fuel sources).

Several international studies have been undertaken to quantify the external costs of nuclear power, i.e. to look at 'externalities', or those effects that are not included in the economic production costs of nuclear power. These externalities may be negative or positive. At a national level, nuclear electricity (like other new renewable sources in national policies against climate change) should be recognized as a low-carbon option, and the tons of CO_2 saved should be considered as positive externalities and evaluated for that. In its study *Nuclear Energy and Addressing Climate Change* (NEA, 2009), the NEA suggests that, in terms of CO_2 emissions/kWh produced in the most modern plants, among the different sources of electricity production, nuclear power emits about 8 g CO_2/kWh, as opposed to 400 g.eq.CO_2/kWh for Combined Cycle Gas Turbine (CCGT) and 1000 g.eq.CO_2/kWh for coal. In this respect, nuclear power, with hydraulics, is an essential tool to reduce base-load electricity CO_2 emissions. The new renewable energies have the same low-carbon characteristics but they cannot be used in base-load production.

Another social or political impact on a national scale which should be considered is the risk of proliferation, even if this is less a risk for a country itself than it is for all other countries (see Chapter 13).

16.2.2 Balancing economic benefits and environmental impacts at a local level

At a local level, the social impacts of the nuclear industry result, on one hand, from the significant economic benefits the industry brings (such as direct and indirect employment, and the building of high value skills) and, on the other hand, from environmental consequences, which are difficult to weigh (such as effects on human health, time required to confine radioactive waste, accident risks, etc.). All these elements are addressed in a very detailed way in the NEA report *Risks and Benefits of Nuclear Energy* (NEA, 2007, pp. 56–73). As the report stresses, there is no consensus on the social impacts of nuclear power, and any indicators considered are partly intuitive and partly resulting from discussion between stakeholders.

In terms of local employment, both direct and indirect employment need to be considered: direct employment during construction (5–10 years), operation (about 60 years) and dismantling (several decades), and indirect employment resulting from local development, notably commercial and education infrastructures, and from the supply chain if it is localized in the country. There are no global statistics regarding local employment resulting from the nuclear industry, and figures can vary greatly from one country to another depending on existing national and local skills, and on the government's and the operator's human resources policy.

To consider the French case, the civil nuclear sector employs about 150,000 people, including about 26,000 EDF (Electricité de France) employees, about 20,000 employees from other companies who work on the maintenance of the 58 plants, and about 55,000 employees of other big companies (Areva, CEA, Andra). To these can be added about 50,000 employees of subcontractors, including those involved in construction, dismantling or maintenance of the plants, and more generally those working for service providers. All branches of engineering are involved, at different levels, including technicians, engineers, researchers, etc. To take the example of EDF's Flamanville site (with two PWR plants in operation – Flamanville 1 and 2 – and one 1600 MWe EPR under construction – Flamanville 3), in 2009, there were 850 permanent jobs (650 EDF, 200 subcontractors), 1800 people working during the plant outages for scheduled maintenance and refuelling, about 40 trainees, and about 100 indirect jobs (trade, catering, security etc.). Construction of Flamanville 3 is scheduled to take place between 2007 and 2014, with 3300 employees on site (40% of whom are local staff, while 60% have been moved in). After 2014, there will be 300 EDF employees on site, 150 subcontractors, and about 900 people for maintenance work during scheduled outages.

The operator has concluded agreements with local communities and local employment organizations, in order to facilitate the gathering of information on local companies, inform employment players of job offers and bids, orientate international and national companies to local employment, and increase local employees' training. There is also a plan to help with retraining after the building process is completed. Indeed, the operators' strong involvement in local development, especially in employment, is the main lever of their public acceptance. This is one of the reasons why it is easier to rebuild a new nuclear plant on an existing nuclear site than it is to find a new site: the nuclear industry is viewed by neighbouring populations as a real asset for local development.

With regard to environmental impacts at the local level, impacts on human health during normal operation of a plant have to be considered, together with the potential effects of major accidents and the time required for radioactive waste confinement.

Several studies have made a comparison between different energy sources regarding the health impacts of normal operation and have shown that nuclear power, along with renewable energies, has the lowest health impact. See, for example, the mortality associated with normal operation of German energy chains in 2000 (Hirschberg *et al.*, 2004). It appears to be clear that nuclear, wind and hydro have the lowest mortality, natural gas and solar photovoltaics are higher, and oil and coal have the highest rate of 'years of life lost'.

The standards for emission of liquid or gaseous effluents include very significant safety margins, so that the human health impacts of a nuclear plant in normal operation are lower than the radioactive emissions found in granite regions, or experienced during a long flight. The standards of authorized emissions have been defined at the international level by UNSCEAR (United Nations Scientific Committee on the Effects of Atomic Radiation). These standards are applied in national regulations. There is a strict control of radioactive emissions from different points of a plant and in its vicinity, and in most countries such data are published by safety authorities and available to the public. Some controversies remain about the low-dose impact of radioactivity on health, which raise epistemological difficulties: how can we prove that there is 'no effect'? All we can do is show that no link has so far been observed between normal emissions and morbidity. Outside normal operation, there have been controversies about the emissions from the Chernobyl accident, and it will take time to assess the emissions from the Fukushima accident and their impacts on the environment. (It needs to be underlined here that the Fukushima accident was a consequence of the combination of an earthquake and a tsunami, and, at the time of writing, there have been no deaths due to radiation in Fukushima. However, the Fukushima accident will of course lead to a global assessment of safety requirements and emergency planning and organization. The Three Mile Island (TMI) accident in 1979, which was a technically severe accident, entailed no health impact on the population.)

Many studies have already been implemented and will yet be implemented to estimate the impact of the Chernobyl accident on human health. It is impossible to make a precise estimation because 'radiation-induced cancers are not all distinguishable from those due to other causes'. And, moreover, other pathologies may also have been caused by radiation. A study published in 2005 by the Chernobyl Forum (an international expert group gathering together several UN agencies including IAEA and UNSCEAR, the World Bank group, Belarus, Ukraine and the Russian Federation) distinguishes three populations exposed to different levels of radiation: 'emergency and recovery operation workers who worked at the Chernobyl power plant and in the exclusion zone after the accident, inhabitants evacuated from contaminated areas, and inhabitants of contaminated areas who were not evacuated.' It concludes that 'the highest doses were received by emergency workers and on-site personnel, in total about 1000 people, during the first days of the accident, ranging from 2 to 20 Gy, which was fatal for some of the workers. Effective doses to the persons evacuated from the Chernobyl accident area in the spring and summer 1986 were estimated to be of the order of 33 mSv on average, with the highest dose of the order of several hundred mSv'. It estimates that 'among the 600,000 persons receiving more significant exposures, the possible increase in cancer mortality due to radiation exposure might be up to a few per cent'. Significant increases of thyroid cancers have been diagnosed among those who were children or adolescents at the time of the accident. This report concludes also that the socio-economic effects of Chernobyl in the contaminated areas should also be as soundly analysed as the health effects. There is no doubt that these are even more difficult to quantify than the health effects.

The social impact of waste confinement, at the local scale, is also a very controversial topic. The 'Not In My Back Yard' (NIMBY) syndrome applies more to waste storage or waste disposal than to nuclear plants, for several reasons. It is difficult to link these facilities to employment, as they do not produce any goods, and employment benefits are limited. Moreover, as will be shown below, a lot of people think that there are no satisfactory solutions for storing High Level Long Life (HLLL) waste, so they fear that a waste disposal plant could entail health consequences for neighbouring inhabitants, and could have a negative impact on the region's image and on local products. Added to this, the time-scale involved with HLLL waste management – millions of years – seems beyond our human comprehension. For philosophical reasons it is very difficult to build confidence about waste management near disposal sites. People think that being given economic compensation is an attempt to buy their acceptance. It seems that strong operator and stakeholder involvement, from the beginning of a project of waste storage or disposal, can ensure better public acceptance of the shared burdens and benefits of steady-priced and cheap electricity. Some interesting experiments in this regard are being implemented in Bure, in north-eastern France, in the area surrounding a geological disposal research laboratory. There, all the radioactive waste producers have been involved in developing local employment opportunities by transferring renewable energy technologies to the area, in parallel with the R&D work being carried out on radioactive waste management. This helps to illustrate the share of responsibility between different regions in French energy policy: the regions which accept radioactive waste disposal benefit from technology transfers to develop also renewable energy sources.

Whatever the technical options considered, it would seem absolutely necessary for newcomer countries to think of a waste management policy right from the moment of the first opportunity study made when launching a nuclear programme, since the waste management question will be raised by their opponents anyway, and then taken up by public opinion at large. It is important to answer public concerns regarding intergenerational responsibility, which is one of the main issues of sustainable development. The goal of such a policy is to avoid passing on unsolvable problems to future generations. Today, several satisfying and secure options exist for managing different categories of radioactive waste (see Chapter 14), including HLLL waste, using geological disposal. The 'problem' of waste management is no longer a technical one but rather a psychological and political issue for local populations.

To conclude this section focused on the 'social impacts' of a nuclear programme, it appears that such impacts are still misunderstood, partly because of an ignorance of scientific matters, partly because of the 'original sin' of nuclear power, and partly because there is no link between statistics concerning risk and intuition, or gut feeling. A better knowledge of the technical and economic facts and figures of nuclear power versus other power sources is a necessary (though not always sufficient) condition to obtain better public acceptance.

16.3 Public perception of nuclear power

The social impacts of a nuclear power choice need to be accepted by the main stakeholders and by public opinion, even if this means allowing the expression of opposition to the policy and, moreover, even if the full meaning of that 'acceptance' is unknown. It needs to be underlined that many people have no real concerns about this topic, except those living near sites, so there is often a 'passive acceptance' among the public at large. It must also be observed that public opinion is very complex, sometimes contradictory or paradoxical or ambivalent about nuclear power, and it is impossible to have a clear understanding of this complexity using only quantitative polls (see below). In any case, there is little 'spontaneous' public perception of nuclear power, except memories of Chernobyl and (since 2011) of Fukushima, and a general link to the atomic bomb, which seems to imply a structural negative image. Beyond this, opinions are built by the media, by political leaders, by a country's political history and its international context, and they can evolve, as is shown by the Swedish case (where there was a referendum with options to phase out the nuclear programme in 1980, but opinion polls in favour of moving to a nuclear programme in 2005, as the international context was increasingly in favour of nuclear power). It is the responsibility of government to give sufficient and honest information on energy to explain and justify the nuclear choice.

An impressive fact, observed everywhere in the world, is that a country's public opinion is more in favour of nuclear energy if a nuclear programme already exists there; similarly, people living in the neighbourhood of nuclear plants are more in favour of nuclear power than the general public.

Nuclear power (and energy in general) is not one of the main concerns of the public at large, except when there is an energy crisis, such as increasing energy prices, or blackouts of supply, or oil spills. All quantitative surveys at a national or international level (for example, the Eurobarometer

Special Report on Energy Technologies (European Commission, 2007a), or the IRSN – Institut pour la Radioprotection et la Sureté Nucléaire – Barometer on perception of risks (IRSN, 2006)) show that social, health and security issues are spontaneously cited as people's main concerns (see Tables 16.2 and 16.3).

It must be observed that, in most surveys, when a question about information on energy is raised, the majority of people (about 70%) say that they

Table 16.2 Responses to the question 'What are the most important issues facing your country today?'

Issue	%
Unemployment	64
Crime	36
Healthcare system	33
Economic situation	30
Immigration	29
Pensions	28
Inflation	26
Education system	19
Terrorism	19
Taxation	19
Housing	15
Energy prices and shortages	14
Environmental protection	12
Public transport	6
Defence and foreign affairs	5

Table 16.3 Responses to the question 'In your opinion, which two of the following should be given priority in your government's energy policy?'

Issue	%
Guaranteeing low prices for consumers	45
Guaranteeing a continuous supply of energy	35
Protecting the environment	29
Protecting public health	22
Guaranteeing your country independence in the field of energy	18
Reducing energy consumption	15
Fighting global warming[a]	13
Guaranteeing the competitiveness of your country's industries	7

[a] Global warming is more and more considered as an important issue but many people still don't know the link between nuclear energy and limiting climate change.

don't have sufficient information about it. However, when public debates are organized, few people from the general public participate in the meetings except those in the neighbourhood of nuclear sites.

We should here consider a number of methodological issues to help us understand public perception of nuclear power. There are many quantitative opinion polls, realized at both a national and an international level, which are very useful to measure a population's degree of knowledge and concern. Eurobarometers, realized under the aegis of the European Commission, are well known and often taken as a reference tool. Such Eurobarometers have addressed different topics regarding nuclear energy: *Europeans and Nuclear Safety Report* (2007b), *Energy Technologies: Knowledge, Perception, Measures* (2007a) and *Radioactive Waste* (2005). The main results of these polls are discussed below, but it is important first to note several limitations of this kind of tool. First, most of the questions raised are closed questions: sometimes the wording doesn't have the same sense for all respondents, and may even be very far from the respondents' concerns. Second, it is worth noting that there are significant differences between European countries, so it is impossible to speak about something like 'European public opinion' regarding nuclear power.

To complement quantitative approaches, qualitative studies (in-depth analyses of people's opinions by non-directive interviews, open questions, etc.) are also useful to have a sound understanding of people's representations, in all their complex and sometimes paradoxical or contradictory aspects. Such a qualitative approach is necessary in order to be aware of all obstacles to nuclear acceptance. For instance, a qualitative study undertaken in France in 2005, before the passing of a new Act on waste management, showed that the public at large were not ready to accept the idea of long-term geological disposal, one of the reference solutions for managing HLLL radioactive waste, because the time-scales involved in waste management (for some categories of waste being as long as a million years) seemed to be, from a philosophical point of view, beyond human responsibility. The appropriate answer was of course not to avoid such a solution in the new Act, but to take into account people's expectations of the reversibility of disposal. The 2006 Act on Sustainable Management of Radioactive Materials and Radioactive Waste requires that any geological disposal be reversible for a minimum of 100 years.

Taking another example, many quantitative polls ask simple questions but phrase them in terms which are not those used by the public in everyday life, or which are difficult to interpret. When discussing nuclear power, questions that are too simple are not relevant. For example, the simple question 'Are you against or in favour of nuclear power?' does not take into account or explain that, in several countries, about 50% of the public have no precise opinion on nuclear power, or have ambivalent perceptions, with some people thinking that 'it is good for the economy and bad for the environment' whilst others think the opposite.

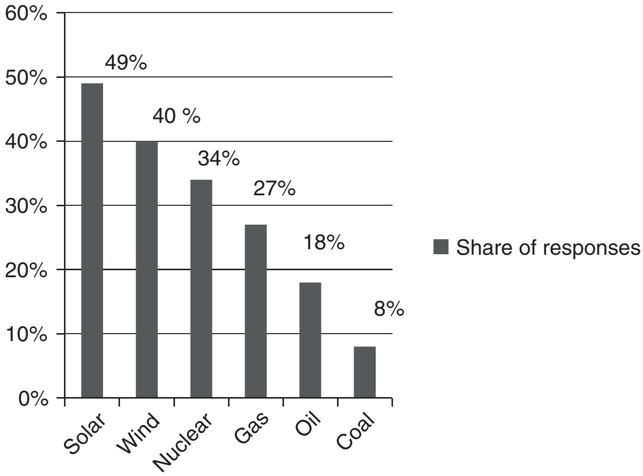
It is also difficult to interpret answers to closed questions such as 'Do you agree/disagree with the following opinion: waste disposal may be implemented in safe conditions', because we do not know what each of the different respondents consider to be 'safe conditions'. The same can be said about the following question asked in the EU Waste Eurobarometer: 'Would you be more in favour of nuclear energy if one would have solved the problem of waste management?', which requires an understanding of what is meant, to the public at large, by having 'solutions' for the problem of waste management; we know that technical solutions already exist but their social acceptance remains problematic.

It therefore seems appropriate to combine quantitative and qualitative approaches: the quantitative approach to have a rough vision of the acceptance and evolution of public opinion, and the qualitative approach to acquire an understanding of people's concerns.

16.3.1 International constants revealed by polls

Whatever the differences measured by polls in different countries (notably in the EU Eurobarometers), there are several data which are commonly observed everywhere:

- People have little knowledge of the share of each energy source, and they tend to overestimate the current share and, moreover, the potential of new renewable energy sources to produce electricity. If, in nuclear countries, many people have a correct perception of the share of nuclear power in current electricity production, they tend to believe that this share, like that of fossil fuels, could be dramatically reduced by 2030 in favour of solar or wind energy sources (see Fig. 16.1).
- There is a correlation between knowledge and acceptance: the more people are informed of the advantages and drawbacks of different power sources, the more nuclear energy is accepted. Notably, when the public knows that nuclear power doesn't emit greenhouse gases and so doesn't contribute to climate change, it is better accepted. In the same way, people who are opposed to nuclear energy think that they are not well informed and tend to consider that there is a lack of information from the nuclear operators and from governments.
- The main arguments in favour of nuclear energy, everywhere, are its contribution to supply security or energy independency, low prices of energy and, increasingly, the fact that nuclear energy does not emit CO_2. The main reasons evoked against nuclear energy are 'waste management problems' and safety risks. In all countries, we find a proportion



16.1 Responses to the question 'What do you expect to be the top three energy sources in 30 years?' (*cf.* Eurobarometer on Energy Technologies, 2007).

of people for nuclear energy, other people against nuclear energy, and an important proportion that are hesitant or without a clear opinion. This group of 'undecided' people is an important target for government information on nuclear energy.

- There are some correlations between socio-demographic characteristics (gender, age, education and economic levels), political opinions and nuclear acceptance. Generally, men are more in favour of nuclear power than women, well-educated people are more in favour of nuclear power than the less well educated, and right-oriented people are more in favour of nuclear power than those who are left-leaning.
- People claim to have more trust in non-governmental organizations (NGOs) and scientists than in political leaders, government and the media to give them information on nuclear power.
- In fact, the public is very much influenced by mass media and by political leaders. There is a vicious circle between the perception by political leaders that nuclear power is not a well-accepted choice by citizens, and that therefore there is some political risk attached to supporting it, and the citizens' perception that there is some reluctance for political leaders to support nuclear power.
- A guarantee of low energy prices for consumers is, everywhere, the main expectation of a government's energy policy. Nevertheless, security of supply, protection of the environment and of human health are also important expectations (see Fig. 16.1).

- Last but not least, public opinion on nuclear power can evolve: a major accident like Chernobyl can have a great impact on nuclear acceptance everywhere in the world; indeed, after Chernobyl, some countries decided to phase out their nuclear programmes, sometimes via a referendum (e.g. Italy). Following the Fukushima accident, German Chancellor Angela Merkel decided to shut down the older nuclear plants in Germany. In this regard, safety is a shared responsibility for all nuclear operators and all nuclear countries. The Fukushima accident will have consequences for nuclear development everywhere, and particularly in Western countries where the nuclear option is more controversial. On the other hand, since the 2000s, there appears to have been more and more acceptance of nuclear power, due to a combination of several factors: progressive awareness of climate change as a major issue, the influence of nuclear development in Asia, signs of a 'nuclear revival' in the EU and USA, the instability of oil and gas prices, geopolitical tensions between suppliers and consumers, the scarcity of raw materials, etc. Making a nuclear choice could be seen as a factor of stability in this context.

16.3.2 Public perception of the radiological risk

It must be remembered that there is always a gap between intuitive perceptions and probabilistic evaluation of risk, in any field: we know that the probability of having a fatal accident when travelling by plane is far lower than having one when travelling by car but, nevertheless, many people are more afraid of being in planes than they are of being in cars. In the energy field, many studies comparing lethal risks resulting from different energy sources (ExternE, NEA, 2010) show that nuclear energy's risk of a lethal accident is lower than that for fossil sources (coal, oil and even gas). Nevertheless, the risk of accident is more spontaneously linked to nuclear power than to coal mining or oil extraction. This risk remains the main argument of nuclear opponents and it is also an obstacle for people who have ambivalent perceptions of nuclear energy.

In the 2007 Eurobarometer, respondents had to choose between two answers: 'The advantages of nuclear power as an energy source outweigh the risks it poses' and 'The risks of nuclear power as an energy source outweigh its advantages' (NEA, 2010: Fig. 2, p. 22). With regard to nuclear power, people's threats are focused on catastrophic accident and radiological risk for human health, often seen as insidious in the neighbourhood of nuclear sites. Objective knowledge may limit fear of these threats, but there always remains some unconscious distrust. But the more people feel well informed on nuclear safety, the less they feel threatened by nuclear safety risks (NEA, 2010, pp. 22–23).

The best way to convince people of nuclear safety is by the example of safe operation: this is why confidence in safety authorities is more pronounced in nuclear countries than in non-nuclear countries and, moreover, more pronounced in the neighbourhood of nuclear plants (Eurobarometer, NEA, 2010, p. 22): 59% of respondents in nuclear countries think that nuclear plants can be operated safely against 31% who do not. This puts the NIMBY syndrome into perspective: opposition particularly applies before the building of a nuclear facility in newcomer countries but is less observed in nuclear countries in the neighbourhood of nuclear plants.

16.3.3 Information, dialogue, debate: how to interact with stakeholders?

In recent years, particularly in western countries, there has been an increasingly marked distinction between giving information, which is a 'top-down' process, and stakeholder involvement, which tends to involve groups and citizens who declare an interest in nuclear choice in the decision process. The degree of citizen involvement in the decision process is variable, ranging from compulsory involvement or (as is more often the case) simply consultative advice. Whichever, stakeholder involvement requires giving sufficient information to stakeholders, and so requires real transparency and access to expertise. However, other information processes may have different purposes: they may have an educational goal, which requires 'objective' information on the advantages and drawbacks of all energy sources, and an explanation of geopolitical and economic constraints which limit and structure energy choices. Conversely, they can tend to involve or influence citizens' opinions, for instance through advertising campaigns, where the goal is less to supply knowledge than to obtain support.

There are many kinds of 'stakeholder involvement processes': national or local public debates; Local Information Councils (CLI), in France, in the neighbourhood of nuclear plants; a Dialogue Forum in the Russian Federation; COWAM in the EU; and numerous other initiatives. These processes don't have the same impact on public decision everywhere: in France, national debates under the aegis of CNDP ('National Commission of Public Debate') have a legal status and are compulsory for some decisions about the building of large energy facilities; again in France, CLIs are compulsory near nuclear sites but they have no decision mandate; in the UK, the 2006 consultation on nuclear policy had no compulsory value; the Swedish process of local consultation to select a disposal site has had a decisive impact on the final choice, etc. The political impact of consultative or participative processes in public decisions to launch a nuclear programme depends strongly on local laws and on national political culture. For instance, in 2005–2006, in France, the government referred to the CNDP to organize a national public debate before passing a new Act on waste management. This public debate was implemented by organizing 13 meetings, some in Paris and in other larger cities (Lyon, Marseille, Nancy) and others in the vicinity of possible waste storage or disposal sites. The schedule was very strict, with the participation of nuclear sector professionals, government representatives, NGOs and independent experts. Some anti-nuclear NGOs refused to participate. Public participation was weak in Paris and in the large cities far from the sites but it was significant near the potential disposal sites. The meetings and an Internet consultation allowed a long list of questions and fears about radioactive waste management to be collected, and for answers to be given to those questions. This also made it possible to take these fears into account when proposing the Act, by including clauses, for example, to ensure the reversibility of nuclear waste disposal for a period of 100 years. This whole process probably increased people's knowledge and understanding of waste management issues in France, but it did not increase the wider public interest in them. Some years later, the question of radioactive waste management remains, for the public at large, a problem with 'no solution'.

The following lessons can be learned from the experience of public debates in France:

- Be transparent about the process and about the role of debate in elaborating a decision; it is important to explain the impact of public debate on the decision (whether about an Act or selection of a site for a nuclear facility). In this regard, several qualitative and quantitative studies conducted in France (IRSN, The French Perception of Risks and Security, Barometer, 2010), and quantitative studies implemented in the EU show that a majority of citizens delegate technological decision to experts, provided the experts report their arguments and possible doubts or disagreements, and provided information is shared with the public. Moreover, the Eurobarometer on Nuclear Safety (2010) showed that 'only around one in four Europeans would like to be directly consulted in the decision-making process regarding the development and updating of energy strategies'.
- Give complete information about all energy sources and allow people to be able to build their own understanding of realistic choices: is there an alternative to nuclear power and, if so, what are the advantages and drawbacks of each alternative solution?
- Clearly define the process and the different steps from opportunity study to decision, and the rules, limits and schedule of public consultation.
- Listen to all the fears and questions raised about a nuclear project and provide answers to all of them.

16.4 Conclusion

A gap remains between the social impact of nuclear power and people's perception of its impact. This gap is more acute in non-nuclear countries, and the more nuclear power is experienced, the better it is accepted. But contradictory phenomena influence the evolution of public perceptions. A better understanding of energy questions and of environmental and economic issues will probably contribute to a better acceptance of nuclear power, and the growing interest in developed and (more and more) developing countries for nuclear power as part of a low-carbon energy strategy may have a driving effect. In the same way, continuous improvements in safety may have a positive effect; conversely, however, a severe accident like Fukushima will have an adverse effect everywhere in the world, even if the needs of nuclear power in an energy mix remain exactly the same as before the accident. There is a growing need for stakeholder involvement in the decision process, which has ambivalent consequences: it can favour objective discussion among different parties about the advantages and drawbacks of the nuclear choice; however, many people in the public at large who have no definite opinion on nuclear power may in fact be convinced by nuclear opponents, of whom there are many giving their views in public debates. As a result, nuclear decisions must be based soundly on a technical and economic opportunity study, and any such decision must be supported by a majority of political decision-makers and by the business community. The most difficult decision to make is the decision for the first plant or the first units, because infrastructure costs and possible public reluctance are the same for one plant or for a whole programme. Such a decision may have a very positive impact on economic, technological, industrial and educational development in a country but requires sufficient political stability to guarantee safe, secure and sustainable practice.

16.5 References and further reading

Bickel, P. and Friedrich, R. (eds) (2005), *ExternE Externalities of Energy Methodology 2005 Update,* European Commission, Directorate General for Research, Sustainable Energy Systems, EUR.

D'Iribarne, P. (2005), *Les Français et les Déchets Nucléaires,* Ministère de l'Industrie, DGEMP, La Documentation Française (http://www.ladocumentationfrancaise.fr).

European Commission (2005), Eurobarometer, *Radioactive Waste,* June (http://ec.europa.eu/energy/nuclear/waste/doc/2005_06_nuclear_waste_en.pdf).

European Commission (2007a), Eurobarometer, *Energy Technologies: Knowledge, Perception, Measures,* January (http://ec.europa.eu/public_opinion/archives/ebs/ebs_262_en.pdf).

European Commission (2007b), Eurobarometer, *Europeans and Nuclear Safety Report*, February (http://ec.europa.eu/public_opinion/archives/ebs/ebs/271_en.pdf).

European Commission (2010), Eurobarometer, *Europeans and Nuclear Safety Report,* March (http://ec.europa.eu/public_opinion/archives/ebs/ebs_324_en.pdf).

Hirschberg, S. R. *et al*. (2004), *Sustainability of Electricity Supply Technologies under German Conditions: A Comparative Evaluation*, PSI report no. 04-15, Paul Scherrer Institute, Willingen, Switzerland.

IAEA (2007), *Milestones in the Development of a National Infrastructure for Nuclear Power*, IAEA Nuclear Energy Series, no. NG-G-3.1, IAEA, Vienna.

IRSN (2006), *Barometer on Perception of Risks* (http://www.irsn.fr).

NEA (2007), *Risks and Benefits of Nuclear Energy*, NEA report no. 6242, Nuclear Energy Agency, Paris.

NEA (2009), *Nuclear Energy and Addressing Climate Change*, Nuclear Energy Agency, Paris (http://www.oecd-nea.org/press/in-perspective/addressing-climate-change.pdf).

NEA (2010), *Public Attitudes to Nuclear Power*, NEA report no. 6859 (http://www.nea.org).

The Chernobyl Forum: 2003–2005. *Chernobyl Legacy: Health, Environmental and Socio-Economic Impacts*, second revised version.

WNA (2010), *The Economics of Nuclear Power*, World Nuclear Association Report, WNA, London (http://www.world-nuclear.org/info/info02.html).

17
Environmental impacts and assessment in nuclear power programmes

I. SALTER, P. ROBINSON, M. FREEMAN and J. JAGASIA, Burges Salmon LLP, UK

Abstract: This chapter analyses how the environmental impacts of nuclear new build are taken into account in government policy, planning decisions and the regulation of plants at all stages in their lifecycles, summarises the legal regime that underpins the requirement for strategic environmental assessments and environmental impact assessments, and considers the key features of land use planning systems and regulatory systems and the role that they play in the control of environmental impacts.

Keywords: strategic environmental assessment, environmental impact assessment, land use planning, environmental regulation, nuclear new build.

17.1 Introduction

The risk of harm to the natural and human environment associated with nuclear installations is undeniably significant, and thus requires proper management. Uncontrolled discharges of radioactive waste will necessarily cause chemical and biological disruption to local ecology and biodiversity, unregulated exposure to radiation is medically proven to pose risks to human health, and the severe environmental consequences of incidents such as Three Mile Island (United States, 1979) and Chernobyl (Soviet Union (Ukraine), 1986) have demonstrated that nuclear operations can have significant implications for internal relations. As a result, there has been long-standing social and political opposition to the development of new nuclear installations on the basis of environmental impacts. The recent events at the Fukushima Daiichi No.1 nuclear plant (Japan, 2011) have served as a stark reminder of the risks associated with nuclear installations, and many countries around the world have paused to reflect on their own national nuclear programmes.

In order to recognise and address these concerns, legal systems, national, supranational and international, have had to develop and evolve suitable processes and mechanisms to ensure not only that the safety of nuclear installations is maximised, but also that the public has the fullest confidence that a thorough consideration of environmental impacts has been fully integrated in the development and planning process. Many jurisdictions now reflect certain internationally accepted legal mechanisms, the primary function of which is to ensure that national public authorities carry out a series of environmental assessments before a decision is taken as to whether to authorise the development of new installations. These assessments will identify the likely environmental impacts of the project, and suggest ways of mitigating these impacts. In addition, the land use planning system is employed in most civil nuclear jurisdictions to decide whether or not new developments should be approved. Amongst other things, planning bodies will take into account the findings of environmental assessments (and other environmental impacts brought to their attention) in their decisions and the resulting conditions that are imposed on successful applicants.

This chapter aims to identify the key procedural and substantive aspects of two types of environmental assessment, and then to demonstrate how environmental impacts associated with nuclear installations are reflected by national planning authorities in their decision making and subsequently regulated throughout the life of an installation.

17.2 Environmental protection

17.2.1 International environmental protection

Ever since the establishment of the United Nations Environment Programme and the formulation of the 1992 Rio Declaration, environmental protection has played an increasingly significant role in international law. Modern international institutions and instruments seek to promote economic development whilst at the same time preventing States from wilfully exploiting or neglecting their natural environments.

One particular area which has been a major focus of the international community, and indeed which dates back to the early part of the twentieth century, is the environmental and human health risks associated with ionising radiation. The first important international institution in this field was the International Commission on Radiological Protection (ICRP), which was established in 1928 to publish recommendations on the basis of scientific research on the risks posed by radiation exposure.

The International Atomic Energy Agency (IAEA) is another important body in this field. As with the ICRP, the IAEA regularly publishes advice and guidance on how national legal systems can best protect individuals and the environment from radiation harm. In 2006, the IAEA published its 2006 Fundamental Safety Principles (IAEA, 2006, SF-1), which is a set of basic principles which should be applied by States to all circumstances which give rise to a radiation risk. The fundamental safety objective is to protect people and the environment from the harmful effects of ionising radiation over the lifetime of a nuclear facility (includes stages/processes such as planning, siting, design, manufacturing, construction, commissioning and operation, as well as decommissioning and closure). This objective should also be applied to the associated activities of transport and management of radioactive material and waste.

The IAEA has also published a highly influential document, *Milestones in the Development of a National Infrastructure for Nuclear Power* (IAEA, 2007), which provides detailed general guidance for States on how to develop national nuclear programmes in an environmentally sensitive manner.

The IAEA is in the process of developing further recommendations and guidance which address the environmental impact of facilities and the environmental consequences of radioactive releases to the natural environment (*Radiological Environmental Impact Analysis for Facilities and Activities* and *Regulatory Control of Radioactive Releases to the Environment from Facilities and Activities*). At the time of publication these safety standards were both under development.

The recommendations and publications of the ICRP and the IAEA have undoubtedly had an enormous influence on the development of the international regulation of nuclear facilities; however, like most instruments of international law, they are not directly enforceable in national legal systems – they are merely published with a view to guiding States on how to best introduce measures to protect individuals and the environment from radiation harm. In order for the measures to apply directly in national legal systems, they must be directly transposed into national law by national legislation.

17.2.2 European Union Directives – supranational and national environmental protection

Following the advent of the EURATOM Treaty (establishing the European Atomic Energy Community), European Union regulation in the field of nuclear installations has historically taken the form of EU Directives. As with international law, EU Directives are not directly applicable in Member States – they must be transposed into the national legal system by national implementing legislation. Two legal concepts have been particularly influential in the regulation of the environmental impacts of nuclear installations – Strategic Environmental Assessment and Environmental Impact Assessment. Both of these legal concepts are discussed in further detail below, as well as the national legislation which implements the relevant EU Directives in the UK.

European Union Strategic Environmental Assessment (SEA)

Strategic Environmental Assessment (SEA) is a mandatory legal requirement in the European Union in respect of plans or programmes which are adopted by EU national public authorities. The SEA regime is a relatively recent concept that is derived from the Directive on strategic environmental assessment (SEA Directive 2001/42/EC), which was transposed into English law by the Environmental Assessment of Plans and Programmes Regulations 2004 (SI 2004/1633). The objective of the Directive is to 'provide for a high level of protection of the environment' by ensuring the 'integration of environmental considerations into the preparation and adoption of plans and programmes' (SEA Directive, 2001, Article 1). 'Plans or programmes' is very widely defined and covers many types of activity, including many kinds of government policy statements.

The SEA Directive provides that an environmental assessment is to be carried out for all plans and programmes which are likely to have significant environmental effects. The assessment is to be completed prior to the plan or programme being adopted so as to ensure that environmental considerations are fully integrated in the process from the outset (SEA Directive, 2001, Article 4(1)), and reasonable alternatives should be identified, described and evaluated where appropriate, taking into account the objectives and geographical scope of the plan or programme (SEA Directive, 2001, Article 4(1)).

Plans and programmes requiring SEA

The SEA Directive and its transposition into the domestic legal systems of Member States of the EU establishes a statutory test to determine whether an SEA assessment is required:

1. Is there a specific legislative, regulatory or administrative requirement for the plan or programme?
2. Does the plan or programme set a framework for future development consents?
3. Is the plan or programme 'likely to have significant environmental effects?'
4. Does the plan or programme relate to a subject matter contemplated by the Directive? Plans or programmes prepared for energy purposes are expressly covered by the SEA Directive and so, in the context of the development of new nuclear power programmes, SEA will often feature as a mandatory requirement. (SEA Directive, 2001, Articles 2(a), 3(1) and 3(4))

A key issue for national, regional and local authorities therefore is to determine (in advance of approval) whether the proposed plan or programme is subject to the requirement of an SEA assessment. This will be important since SEA carries with it certain minimum administrative and procedural steps and requirements for consultation that lead to the production of formal documents such as the Environmental Report (akin to the Environmental Statement in a project-level Environmental Impact Assessment – see below). In the United Kingdom, a local authority's decision may be challenged by way of judicial review if, for example, it incorrectly determines that an SEA is not required.

The United Kingdom government has taken the view that pure statements of general government policy do not fall within the scope of the SEA Directive, such as the Energy White Papers of 2006 (Department of Trade and Industry, *The Energy Challenge: Energy Review Report*, July 2006) or 2008 (Department for Business, Enterprise and Regulatory Reform, *Meeting the Energy Challenge: A White Paper on Nuclear Power*, January 2008), and so do not require an SEA assessment to be carried out in advance of their adoption or publication. This is for a number of reasons, the principal one being that the preparation of policy documentation is not specifically required by legislation or a mandatory administrative process. So how does an authority determine whether its proposed 'plan or programme' is subject to the requirements of the SEA Directive? The answer must lie, to a certain extent, in the SEA Directive itself, but particular consideration must also be given to what, in practice, the drafting of policy documents will lead to. Is the document one that is specifically required by legislation? Will it be used as a framework (or part of a framework) for subsequent development consent decisions? Are the issues it addresses ones that are likely to have significant environmental effects? If the plan or programme is to proceed on the basis of identifying development suitability on a site-specific basis, there will inevitably be greater pressure to ensure that SEA is undertaken.

Take the proposed United Kingdom Nuclear National Policy Statement (NPS) as an example; all of these criteria are clearly met. Having additionally resolved to invite the nomination of specific sites for assessment against a range of criteria relevant to the subsequent grant of a development consent for new nuclear power stations, the United Kingdom government has accepted it is inevitable that the SEA process must be adhered to. For the nuclear NPS, the UK Department of Energy and Climate Change/ Office for Nuclear Development (OND) has indicated that an 'Assessment of Sustainability' (AoS) will be undertaken that discharges all the requirements of the SEA Regulations. For this Nuclear NPS, at least, the AoS may replace SEA (see '*Towards a Nuclear National Policy Statement*', OND, January 2009). Although it is now undertaking that process at the same time as drafting the NPS, the UK government recognises that formal stages of SEA are such that the draft NPS and the formal SEA Environmental

Report cannot be one and the same thing, but have to be offset. The NPS must have been issued in draft before the Environmental Report under the SEA Regulations can be prepared.

So, in cases such as this, policy makers are faced with the challenge of ensuring that there is adequate environmental investigation of the policy that they intend to put forward before the policy is formalised. In this respect, the United Kingdom's forthcoming Nuclear NPS is expected to set the standard for the level of information required for the drafting of an NPS that is site-specific.

Practical application and key considerations

The primary output of the SEA process is the Environmental Report, a document required by the SEA Directive which identifies the likely significant effects on the environment that would occur if the plan or programme were to be implemented (SEA Directive, 2001, Article 5(1)). Annex I of the SEA Directive sets out the minimum information that the Environmental Report should contain. Given that the SEA process is essentially a comparison of the state of the natural and human environment with and without the plan or programme being implemented, the starting point of the Environmental Report is to identify the current state of the environment and how that area would evolve without implementation of the plan or programme (SEA Directive, 2001, Annex 1, paragraph (b)). The Environmental Report should also identify the broad environmental characteristics of the area likely to be affected, as well as the particular areas where impacts could be significant, such as biodiversity, population, human health, archaeological heritage and landscape (SEA Directive, 2001, Annex 1, paragraphs (c) and (f)).

Perhaps the most significant part of the Environmental Report, however, at least in terms of the ideological drive behind the SEA Directive, is the part which identifies the measures envisaged to 'prevent, reduce and as fully as possible offset' (SEA Directive, 2001, Annex 1, paragraph (g)) the significant adverse impact on the environment. Indeed, the SEA process would have very little worth if authorities were not obliged to consider ways to mitigate the serious environmental effects that have been identified by the process. These measures should be drawn up in light of current knowledge and methods of assessment and the contents and level of detail in the plan or programme so as to ensure that the local authority can identify mitigation measures which would be commensurate with the nature and extent of the likely environmental effects (SEA Directive, 2001, Article 5(2)).

As well as identifying practical measures that would mitigate the environmental impact of the proposed plan or programme, the SEA Directive also requires the Environmental Report to identify, describe and evaluate the 'reasonable alternatives taking into account the objectives and the geographical scope of the plan or programme' (SEA Directive, 2001, Article 5(1)). This requirement is designed to ensure that the authority gives serious consideration to the environmental impacts of the proposed activity and applies its mind to alternative plans or programmes (or variations on the existing plan or programme) what would have less serious consequences for the environment. On a practical level, the area of alternatives is often the most closely scrutinised by interested parties and, in the United Kingdom in particular, local authorities have been particularly cautious in their approach to identifying alternatives.

Links between SEA and EIA

Despite the fact that SEA has been a legal requirement since 2005, there has been relatively little commentary or practice that has emerged on what authorities are required to do with the output of the SEA process. As a result, this area is relatively untested in the context of major infrastructure development, and particularly in the area of nuclear/energy planning. The key question remains exactly how, if at all, local authorities should aim to integrate the results of the SEA process, and the conclusions of the Environmental Report, with the subsequent process of granting individual development consents for projects.

Even outside the energy sector there is surprisingly little written guidance on this and indeed very little written legal opinion on the appropriate use of SEA materials at the development consent stage (such challenges relating to SEA that we have seen relating to the policies themselves, not the subsequent reliance upon them). This has the somewhat unfortunate result of potentially leaving authorities with the mistaken belief that the SEA process is an exercise with no real end. Nonetheless, this is not a safe assumption to make and does not sit at all comfortably with the express objective of the SEA Directive to ensure that environmental considerations are fully integrated in the development process. The way in which the SEA output material influences project-level development control decisions will be how, in practice, the process of SEA influences development on the ground with a view to promoting sustainable development.

In principle, the answer to this problem is relatively simple – when a developer makes a project-specific development consent application which relies upon, or perhaps more widely just bears upon, any 'plan or programme' which has been subject to the SEA regime, it will be important for the determining authority to at least make references to the Environmental Report, the output of the SEA regime which should have already been completed. The usual method by which this is effected is through the detailed project-level Environmental Impact Assessment

(EIA). There are undoubtedly some areas of overlap between the SEA process and the EIA process, but the Commission of the European Union has broadly distinguished the two on the basis that SEA applies 'upstream' to certain public plans and programmes, while EIA applies 'downstream' to certain public and private projects (European Commission, 2009, paragraph 3.5).

17.3 Environmental Impact Assessment (EIA)

17.3.1 Background to European Union Environmental Impact Assessment

The Environmental Impact Assessment (EIA), also known as the Enviornmental Impact Statement in the United States of America, is 'an examination, analysis and assessment of planned activities with a view to ensuring environmentally sound and sustainable development' (United Nations, 1987). Although the definition of EIA does appear strikingly similar to that of SEA, the fundamental distinction between EIA and SEA is essentially one of tiering: SEA is carried out at an early stage to assess the environmental impacts of a proposed plan or programme; EIA is carried out at a later stage in the development process when the authority has undertaken the SEA process and is considering granting development consents for a specific development activity. Specifically in the context of the development of new nuclear programmes, the approach to EIA can be contrasted with the high-level approach to regulatory justification required by ICRP 60 (International Commission on Radiological Protection, 1990) and other legislative instruments developed within the European Union (see EC Directive 96/29/EURATOM) and transposed into European Union Member States (for example, SI 2004/1769 on nuclear justification in the UK). EIA is a detailed, project-specific assessment of the environmental impacts of a proposed project.

Notwithstanding the fact that the EIA process comes after that of SEA, it remains paramount that EIA is undertaken at a very early stage in the decision-making process, crucially before a decision is taken as to whether consent for the development should be granted. Relevant significant environmental issues should be identified and impartially examined, so that national authorities do not undertake or authorise the activities in question without serious prior consideration of their environmental impacts. To this end, EIA is a necessary legislative tool in any regulatory system which aims to promote a certain level of concern between economic development and environmental protection.

It is commonly accepted that the concept of EIA has its earliest roots in legislation from the United States, the National Environmental Policy Act

1969 (NEPA), which was passed largely in response to the public's heightened concern for the environment raised by Rachel Carson's *Silent Spring*. The express purpose of NEPA was to 'promote efforts which will prevent or eliminate damage to the environment' by establishing 'a national policy which will encourage productive and enjoyable harmony between man and his environment' (NEPA, Section 2). NEPA established a legal mechanism whereby federal agencies were compelled to prepare a 'detailed statement' of the environmental impacts of proposed projects, a statement which became known as an Environmental Impact Statement, and to 'study, develop, and describe appropriate alternatives' to the proposed course of action (NEPA, Sections 102(2)(C) and (E)). The process under NEPA bears many similarities with the modern-day SEA and EIA processes, principally in that it aims to compel the institutionalisation of environmental concern, and to ensure that the views of a wide range of parties, including the public, are incorporated in the decision-making process.

Since the 1960s, the principles established by NEPA have been refined and developed and are now enshrined at an international level in a number of legal instruments. In 1987, the United Nations Environment Programme demonstrated its support for the concept of EIA through the publication of its 'Goals and Principles of Environmental Impact Assessment' (United Nations Environment Programme, 1987), a comprehensive overview of EIA methodology at national, regional and international levels. Further support was given to EIA by Principle 17 of the Rio Declaration which advocates the use of EIA as 'a national instrument' for 'proposed activities that are likely to have a significant adverse impact on the environment' (United Nations, 1992, Principle 17). The European Union has also passed several Directives requiring Member States to legislate for the assessment of the environmental effects of public and private projects, the most notable in this area being Directive 85/337/EEC (the EIA Directive) which, in general, has been transposed and implemented in all Member States.

It is clear, then, that the concept of EIA is widely accepted by the international legal community, principally on the basis that the process should introduce a certain level of impartiality, transparency and accountability to decisions that will necessarily have a significant impact on the natural and human environment. EIA also provides a valuable opening for public participation in decision-making, even though public opinion will not necessarily prevent the project from proceeding. The United Kingdom House of Lords has held that the obligation on authorities is to ensure that the EIA process is an 'inclusive and democratic procedure . . . in which the public, however misguided or wrongheaded its views may be, is given an opportunity to express its opinion on the environmental issues' (*Berkeley v Secretary of State for the Environment and Another* [2001]). Public participation is a fundamental tenet of the EIA Directive regime, and is also a factor which becomes particularly poignant when considered in light of the obligations of many States under the Aarhus Convention (Aarhus, 1998) and certain international human rights agreements (such as the European Convention on Human Rights and Fundamental Freedoms). Specifically in the context of nuclear new build, the IAEA International Nuclear Safety Group (INSAG) has emphasised the importance of public participation as a way of ensuring public confidence in the safety of nuclear installations (INSAG, 2006).

17.3.2 Processes requiring EIA

It is clear from Principle 17 of the Rio Declaration, in addition to other international environmental instruments, that international law requires States to carry out an EIA when the proposed project is considered to have 'significant adverse' implications for the environment. A 'project' is generally accepted to be the execution of construction works or other interventions in the natural surroundings and landscape (EIA Directive, 2009, Article 1(2)). The obligation to undertake an EIA will not apply where the potential environmental harm that may be caused by the project is slight. Therefore, the potential environmental harm must be significant. The legal terminology is similar at a European level, with EU Member States being required to carry out an assessment where a proposed project is likely to have 'significant effects' on the environment. It remains, however, the discretion of the individual State to determine when the potential environmental impacts can be classed as 'significant'. A key question, therefore, is whether there are parameters to this discretion – are there any types of activity that will necessarily cause environmental damage of the requisite significance so as to require a mandatory EIA?

The EIA Directive is of some assistance on this point. Annex 1 of the EIA Directive lists projects which, by their very nature, must be made subject to a mandatory EIA. 'Nuclear power stations and other nuclear reactors' and 'installations solely designed for the permanent storage or final disposal of radioactive waste' are expressly listed in Annex 1 and so development projects of this nature will always trigger the requirement for mandatory EIA. The European Court of Justice (ECJ) has recently ruled that Annex 1 activities carry with them a presumption of significant environmental damage or risk and therefore will always be subject to the 'unequivocal obligation' (Case C-431/92 *Commission v Germany*, paragraph 39) to carry out EIA, irrespective of whether the activity in question crosses the political boundaries of two or more Member States (Case C-205/08 *Umweltanwalt von Kaernten*).

17.3.3 Transboundary harm from hazardous activities

This raises the interesting question of how EIA should be carried out where environmental consequences of a proposed nuclear installation may straddle political boundaries. Of particular relevance here is the concept of transboundary EIA, the process whereby States 'take all appropriate and effective measures to prevent, reduce and control significant adverse transboundary environmental impact from proposed activities' (United Nations, 1991, Article 2(1)). This is an express requirement of the EIA Directive, but is also an established principle of international law. The Convention on Environmental Impact Assessment in a Transboundary Context (the Transboundary EIA Convention) (United Nations, 1991) is the primary international agreement on the matter, and provided the basis upon which the EIA Directive was amended in 1997. The Transboundary EIA Convention has as its core the objective of enhancing international cooperation in assessing and mitigating environmental impacts in a transboundary context, and addresses the situation where an activity proposed in a territory in one jurisdiction causes the risk of significant adverse environmental impacts in the jurisdiction of another State. The definition of 'impact' is drafted widely and includes 'any effect on the environment . . . historical monuments or other physical structures' and any resulting 'effects on cultural heritage or socioeconomic conditions' (United Nations, 1991, Article 1(vii)). Clearly, this threshold is deliberately set at a very low level so as to encourage active dialogue between States when the potential for transboundary environmental issues arises.

The process affords the affected State the right to participate in and, to a limited extent, influence the decision-making process in the State where the activity is proposed, principally by giving the affected State the right to be notified of the proposed activity and to receive certain documentation as regards environmental assessment. As with EIA at a national level, however, the transboundary EIA process does not give affected States(s) the right to 'veto' a proposed activity on the basis of transboundary environmental impacts.

Adopting an approach similar to that of the EIA Directive, Appendix 1 of the Transboundary EIA Convention identifies types of projects for which a transboundary EIA should always be carried out. This includes proposed development of installations for the 'production or enrichment of nuclear fuels, for the reprocessing of irradiated nuclear fuels or for the storage, disposal and processing of radioactive waste' (United Nations, 1991, Appendix 1, paragraph 3). In the early to mid-1990s, the British government listened closely to representations made by the Irish government when it was considering how to proceed with determining a licence application for a proposed nuclear waste disposal site at Sellafield in England. The Irish government produced strong scientific evidence that the storage of radioactive substances at the proposed disposal site could have significant adverse environmental impacts in Ireland, primarily as a result of Sellafield's geographical location on the Irish Sea coastline. This evidence ultimately played a significant part in the British government's decision to not grant the licence.

17.3.4 Administration of EIA

The EIA Directive requires the production of an Environmental Statement as the primary output of the EIA process, the minimum contents of which are prescribed by the EIA Directive and are closely aligned to that of the Environmental Report in the SEA process. Annex III of the EIA Directive dictates the minimum information that is to be provided as part of the Environmental Statement including, among others, a description of the physical characteristics of the project including its land-use requirements, an estimate (by type and quantity) of expected residues and emissions associated with the activity and a description of the likely significant effects of the proposed activity on the environment. A non-technical summary of the information is also to be included in the Environmental Statement so as to ensure that the implications of the scientific information are readily accessible by the general public. As with the SEA process, a crucial requirement of the EIA Directive is that the Environmental Statement identifies any measures envisaged to 'prevent, reduce and where possible offset any significant adverse effects on the environment' (EIA Directive, Annex 3, paragraph 5). The Environmental Statement is to be made available to the relevant members of the public, along with the application for development consent, and the public is to be given the opportunity to express its opinion on the project before any decision to initiate the project is taken.

However burdensome the EIA process may appear, one fundamental factor (and some may argue flaw) in the process is that an Environmental Statement which suggests significant harm to the environment does not actually prevent an authority from granting its consent for the activity in question. While the EIA Directive expressly requires the decision-maker to take the findings of the Environmental Statement into consideration, there is no overt obligation on the decision-maker to withhold consent for development where the negative environmental effects appear disproportionately greater than the benefits that the activity would bring. Equally, neither is there an obligation on authorities to afford particular weight to the views of the public – although the public has the right to be consulted during the process, the practical value of that right is merely procedural.

Nonetheless, the authority must inform the public of its ultimate decision as well as the reasons and considerations upon which it is based.

17.3.5 Practical considerations

A key practical consideration in relation to EIA is who should be responsible for carrying out the EIA assessment and producing the Environmental Statement, and thus bear the costs of doing so. The EIA Directive stipulates that the developer is to carry out the EIA and provide the requisite information to the authority. For the purposes of the EIA directive, the 'developer' is defined as either the person making an application for authorisation of a private project, or the public authority which initiates a project. Accordingly, the obligation to carry out the EIA lies firmly with the party initiating the project, and in the case where this is a private party, the obligation on the authority is to ensure that the EIA has been properly formulated. This will necessarily mean that early engagement with the process is essential, for both the developer and the authority, particularly so that detailed arrangements for public consultation can be coordinated.

However, the EIA process does not (and should not) end with the decision of the authority giving consent for the activity to proceed. Where an activity is deemed to be justified in light of its environmental effects, the activity and its effects on the environment should be subject to appropriate supervision. This process is known as 'monitoring' and can be distinguished from the main EIA process and preparation of the Environmental Statement on the basis that it should continue throughout the life of the activity in question. The purpose of monitoring is essentially to ensure that the environmental effects which were identified in the EIA Environmental Statement were correct, but also to provide authorities with sufficient information to enable them to decide whether enhanced measures are required to mitigate the environmental damage that will occur. This additional facet of EIA is not necessarily present in all Member States' domestic legislation. In the UK, for example, a development consent granted in reliance on EIA will usually have conditions attached where these are seen as necessary to ensure that environmental impacts are no greater than predicted. However, the Environmental Statement does not, of itself, create an enforceable set of standards to be applied to the development. In a nuclear context, monitoring will necessarily extend beyond the life of a nuclear power plant, and continue throughout the decommissioning phase, with the primary purpose being to ensure that any hazardous or radioactive substances remaining on the nuclear-licensed site do not cause material harm to the natural environment. In the United Kingdom, a separate, comprehensive EIA procedure must be complied with before the process of dismantling or decommissioning a nuclear reactor can commence.

Under the EIA Directive, EU Member States are also required to engage in dialogue with the European Commission for the purposes of exchanging information on the experience gained in applying the EIA process. The reasoning behind this obligation may have had something to do with the discretion that Member States are afforded in setting the threshold of 'significance' in determining whether a proposed Annex II activity requires an EIA. This would seem to be supported by the EIA Directive, which further requires that Member States inform the European Commission of any criteria and/or thresholds adopted for Annex II projects, so as to ensure relative harmonisation of EIA standards across Member States. This is no doubt a fundamental, but secondary, requirement of the EIA Directive, since it does not actually establish any obligatory environmental standards that must be adhered to – quality control is largely a matter for individual Member States. Nonetheless, the European experience of EIA has shown that it is a valuable and successful tool in ensuring that a national planning system adequately addresses and adapts to environmental concerns.

17.4 Land planning for new nuclear

Although the particulars will vary considerably between jurisdictions, the decision to build a new nuclear plant will invariably involve a consideration of the anticipated environmental impacts, and whether they can be mitigated, or even tolerated, to a level whereby the benefits of the development will sufficiently outweigh them. This is not a trivial consideration and history has demonstrated the role that environmental considerations can play in shaping development decisions. There are a variety of different stakeholders involved that can influence the decision-making process in favour of the environment such as concerned local residents, nongovernmental organisations and the environmental and planning authorities. All of these interested groups have the potential to ensure that the environmental impacts of a proposed development, taking into account the information available, including that generated from environmental assessments, are factored into the decision on whether or not to proceed. These considerations will also have an effect on the resulting site licensing conditions if the authorities ultimately do provide consent for a proposed installation.

In most civil nuclear States, the planning system is employed to manage the process and help ensure that there has been adequate scrutiny of environmental impacts. Environmental considerations play a pervasive role in the planning debate and there is therefore a need to understand its processes and the key authorities involved. Although there are common features which unite the legal processes of individual States, this is an area where they have retained considerable autonomy. There are, therefore, a number of permutations and no approach to draw from which will be generally applicable.

For example, many of the relevant provisions that govern the nuclear planning regime in the US are contained in Part 51 of the United States Regulatory Commission's NRC Regulations (10 CFR Part 51). In addition, IAEA documentation such as the *Fundamental Safety Principles* (IAEA, no. SF-1), *Milestones in the Development of a National Infrastructure for Nuclear Power* (IAEA, no. NG-G-3.1) and *Stakeholder Involvement in Nuclear Issues* (IAEA, INSAG-20) are all important starting points from which most civil nuclear States have developed their domestic legal systems. Although there is no common approach, the UK experience serves as an instructive model with respect to land-use planning. The Planning Act 2008 (PA 2008) has introduced significant changes which will alter the content and procedure of the planning process (albeit subject to further change following the election of the coalition government in the UK in May 2010). Added to this is the considerable international interest in the UK market relating to its commitment to a new generation of nuclear power stations. There is significant common ground between the planning system in the UK and other civil nuclear jurisdictions, and a general understanding of its key features will be beneficial to a variety of different stakeholders.

17.4.1 Who is involved?

Planning authorities

All decisions to build new plant will require review by a competent authority in the affected jurisdiction. This is recognised by the IAEA as a fundamental feature of nuclear law and commonly referred to as the permission principle. Their *Handbook on Nuclear Law* describes the principle in the following terms:

> . . . this principle holds that, unless specifically exempted, any activity related to the use of nuclear material and technology should be permitted only after competent authorities have determined that it can be conducted in a manner that does not pose an unacceptable risk to public health, safety and the environment . . . Where a nuclear related activity is deemed to pose a significant health or safety risk, governments require that an explicit authorization be issued by the regulatory body following an application and review process . . . The national legal infrastructure in each State will determine the conditions and procedures applicable to such authorizations and notifications, including any limits on the regulatory body's power to impose additional requirements (Stoiber *et al.*, 2003, pp. 34–35).

Prior to 2009, the UK planning system required the consent of the Secretary of State for the construction of any form of power station with a capacity greater than 50 megawatts. This was a requirement imposed by Section 36 of the Electricity Act 1989. The grant of consent operated in such a way that the applicant was usually deemed to have also been given planning permission (see Town and Country Planning Act 1990 (TCPA, 1990, section 90(2)). Although to a certain extent the discretion of the Secretary of State was fettered by the evidence presented (including that from EIAs and public inquiries) which had to be judged against set criteria, the powers provided were extensive.

The Section 36 consenting procedure no longer applies in the UK and it has been replaced by a new regime introduced by the PA 2008 (McCracken, 2009). It was widely felt that the Section 36 regime was unsuitable for consenting major infrastructure projects and too time-consuming. The challenge was 'to transform the regime for major infrastructure projects in order to achieve outcomes that are both faster and fairer; both more efficient and more accountable; and which both ensure more timely delivery, and improve the ability of communities and individuals to participate in the system' (Kelly, 2008, p. 2). The Infrastructure Planning Commission (IPC) was created to decide applications relating to 'nationally significantly infrastructure' such as generating stations, highways, airports, railways and hazardous waste facilities. Within a few months of its formation, the coalition government decided to abolish the IPC and replace it with another new body called the Major Planning Infrastructure Unit (MPIU) which will operate as a specialised branch of the Planning Inspectorate. The key reason for this change was that the coalition government wanted to ensure that elected ministers would be vested with decision-making powers rather than unelected IPC commissioners. At the time of publication, the legislation which is intended to replace the IPC has not been given effect and the IPC continues to be the relevant decision-making body in the intervening period. Despite the impending reform, it is expected that most of the changes introduced by the PA 2008 will be retained going forwards. The 50 megawatts threshold continues to apply to generating stations, which effectively means that all new nuclear power plants will have to seek development consent from the IPC/MPIU. Schedule 1 of the PA 2008 fleshes out important constitutional details of the IPC and provides the Secretary of State with the powers to appoint the Commissioners. The creation of a specialist body such as the IPC/MPIU requires the appointment of Commissioners/Ministers with the necessary expertise to assess major development proposals. Although the IPC (until it is replaced by the MPIU) is vested with most of the power to determine applications falling within their remit, the Secretary of State has retained residual powers (Sections 110–113 of the PA 2008) to intervene in the interests of defence or national security. The new planning regime in the UK shifts power from the government to the IPC/MPIU, but these steps towards independence have been offset by a suite of measures that have been introduced to ensure parliamentary accountability (Tromans, 2010, p. 141).

Regulators

The regulators will have an ancillary role to play in shaping the land-use planning debate. The authority tasked with overall responsibility for the regulation of nuclear installations will generally be involved in key planning decisions, since they will bear most of the regulatory responsibility for the plant during its lifetime. In the UK, this function is performed by the Nuclear Directorate (ND), a specialist organisation within the Health and Safety Executive (HSE), responsible for setting, monitoring and enforcing safety and security standards on nuclear sites. There are a number of other regulators that will have an interest in the planning debate, such as the Environment Agency (EA), the Office for Civil Nuclear Security, the Department for Transport and the coastal authorities. It is imperative for applicants to engage with the regulators from the outset because the IPC/MPIU will 'expect the applicant to have involved the relevant regulators at the pre-application stage so that the applicant can incorporate the regulators' requirements in proposals' (draft Nuclear National Policy Statement EN-6, paragraph 3.4.4).

17.4.2 Permissions and processes required

IAEA milestones

The IAEA document entitled *Milestones in the Development of a National Infrastructure for Nuclear Power* (IAEA, no. NG-G-3.1) sets out a framework of milestones in the development of a national nuclear infrastructure. The three core milestones are: (1) ready to make a knowledgeable commitment to a nuclear programme; (2) ready to invite bids for the first nuclear power plant; and (3) ready to commission and operate the first nuclear power plant. The first of these milestones is of paramount importance whether a State is deciding to embark on its first nuclear power plant or, as in the UK, deciding to commission a new fleet of nuclear power plants. If there is no desire on the part of the legislature or national authorities in a given State, installations will not generate enough interest to survive to the planning stages of a development. In the UK, after many years of indecision, milestone 1 was achieved when government energy policy, enshrined in legislative White Papers, made an express commitment to a new generation of nuclear plants.

National policy statements (NPS)

Part 2 of the PA 2008 empowers the Secretary of State to publish NPS, following Parliamentary scrutiny, in relation to specified descriptions of development. The introduction of NPS is part of the strategy to expedite planning timeframes. By formalising government policy in advance in an overarching document, NPS are meant to avoid policy disputes being raised further down the line in respect of specific project applications. In November 2009, the UK government published a draft overarching NPS for energy (EN-1) and a draft NPS specific to nuclear power generation (EN-6) which, at the time of publication, are still in draft form. The consideration of environmental impacts figures prominently in both draft NPS and certain core areas have been specifically highlighted in the context of nuclear development such as flood risk, water quality and resources, coastal change and biodiversity and geological conservation (EN-6, p. 27). These impacts will need to be addressed by applicants in their environmental assessments and the IPC is obliged to ensure that they have been adequately factored into their decision-making. Another way in which environmental impacts have been taken into account in the draft nuclear NPS is in the siting of new power stations. Part 5 of the NPS identifies 10 potentially suitable sites for new development. The assessment criteria included a consideration of certain environmental impacts, and the sites were chosen following consultation with the UK environment agencies.

Planning procedure

Consultation

This will be an important feature of most land-use planning systems and provides interested parties with the opportunity to influence the outcome of planning decisions. Part 5 of the PA 2008 has fundamentally reformed the way in which the UK approaches consultation and imposes a substantial burden on applicants to consult with affected communities and the local authorities. The onus is on applicants to be satisfied that they have discharged their consultation obligations at the pre-application stage. This is a measure which was introduced to improve the efficiency of the planning system, and it is hoped that front-loading the consultation exercise will enable problems, including local environmental impacts, to be identified at an earlier stage in the process. Applicants are required to take account of the responses to consultation when deciding whether to proceed with a given project, and the IPC/MPIU will eventually be provided with a copy of a consultation report detailing what has been done to consult, any relevant responses and how they have been accounted for. Section 55(4) of the PA 2008 obliges the IPC to have regard to the report when making their decisions.

The application

Regulation 5 of the Infrastructure Planning (Applications: Prescribed Forms and Procedure) Regulations 2009 prescribes the various documents and information that must accompany applications for orders granting development consent. Significantly, this includes the production of an environmental statement (and any relevant scoping and screening opinions) pursuant to the Infrastructure (Environmental Impact Assessment) Regulations 2009. The IPC/MPIU is compelled by Regulation 3(2) and (3) not to grant development consent unless it has first taken the environmental information into consideration and stated in its decision that it has done so. The requirement to furnish an environmental statement with the application obliges applicants to advertise and consult on the environmental statement at the pre-application stage in conjunction with their general consulting obligations. In a similar vein, Part 51 of the United States Regulatory Commission's NRC Regulations (10 CFR Part 51) also prescribes detailed information on the requirements and content of environmental impact assessments for US new build.

Section 55(2) of the PA 2008 requires the IPC/MPIU to decide whether or not to accept applications for further examination within 28 days of their receipt. Once accepted, the IPC/MPIU must invite affected local authorities to submit local impact reports detailing the likely impact of the proposed development on the authority's area. This is another opportunity for the environmental impacts to be considered and local issues adequately addressed. As Tromans notes (2010, p. 145), 'there will be an important relationship to be worked out between the local impact report and the environmental statement produced by . . . the applicant. The impact report may present a quite different perspective on what are regarded as likely and significant impacts'.

Examination and decision

The examination process is not intended to take more than six months, and Section 88 of the PA 2008 requires the examining authority of the IPC/MPIU, after an initial assessment of the issues, to meet with the applicant and each other interested party to allow them to make representations about how the application should be examined. Written representations will be the norm, and this marks a big change from the previous regime under Section 36 of the Electricity Act 1989 (or its predecessors) which was characterised by potentially long and expensive public inquiries (e.g. the Sizewell B inquiry which lasted for 340 days). Cross-examination under the new system will play a more limited role, although interested parties do have the opportunity to make representations in open-floor hearings. The IPC does, however, have considerable powers to control the content and procedure of these hearings.

Unless extended by the Chair of the IPC/MPIU or the Secretary of State, the decision must be made within three months of the completion of the examination stage. The application must be decided in accordance with the nuclear-specific NPS, unless one of five exemptions applies, including where the panel is satisfied that the adverse impact of the proposed development would outweigh its benefits. Decisions must be supported by a statement of reasons and, if granted, the IPC/MPIU can impose conditions in the order granting development consent.

17.4.3 Key considerations

Challenge

The legal systems of most jurisdictions will provide mechanisms for the challenging of planning decisions because they are an important constitutional feature of a developed legal system in a democratic State. In the UK, such decisions are challengeable by judicial review proceedings in the High Court and the PA 2008 requires them to be initiated within six weeks of the publication of the reasons for the decision. The court will intervene only in limited circumstances if the planning decision is illegal or unreasonable or if there has been procedural unfairness. Planning decisions can be challenged by parties with sufficient standing, which includes many opponents of nuclear power and also unsuccessful applicants. The long history of environmental opposition to nuclear power from NGOs makes the prospects of legal challenge likely and promoters will need to plan for this possibility.

Community benefit

The planning systems of most States will provide some form of community benefit mechanism whereby promoters can suggest (or be required to provide) social benefits to the communities that will be affected by the siting of the nuclear installation. The Finnish experience is instructive and their system of community benefits played a key role in helping to overcome public opposition to the Onkala geological disposal waste repository. Like other jurisdictions, UK planning law allows applicants to enter into binding agreements with local communities to provide benefits in recognition of the burden that they shoulder on behalf of wider regional and national interests. Community benefits are increasingly viewed as an integral part of the public engagement process. Applicants will need to consider their provision at an early stage. These funding commitments also existed under the old consenting regime as well. Whilst the discretion to offer such benefits is wide, the ability of local authorities to demand such arrangements is narrower. Government guidance (paragraph B5 of Circular 05/2005) sets out a number of tests which must be met before planning obligations can be required by local authorities and provides that they 'must be relevant to planning; necessary to make the proposed development acceptable in planning terms; directly related to the proposed development; fairly and reasonably related in scale and kind to the proposed development; and reasonable in all other respects'. Although these obligations operate *vis-à-vis* local authorities, they will no doubt be an important influence on IPC decision-making.

Key environmental impacts within the nuclear context

The draft nuclear NPS identifies certain nuclear-specific environmental impacts which will need to be appropriately addressed in the EIAs which will accompany applications.

Water quality and resources

The construction of a nuclear power plant can adversely affect the water quality and resources in the area through increased demand, the thermal impact of cooling water discharges and the disruption of designated habitats of ecological importance. Where these impacts are likely, applicants will need to ensure that their EIA identifies the existing levels of water quality, discharges and abstraction within the area, and the cumulative effect when considered with other industrial sites or projects existing or planned. Applicants are also required to set out the characteristics of cooling water for new nuclear power stations and the implications on marine and estuarine environments, and will be expected to mitigate the hydrologic impacts of their activities.

Coastal change and marine impact

The requirement in the UK to site nuclear power stations in coastal locations can have an environmental impact on marine processes. Paragraph 4.4.1 of the draft NPS recognises this and provides that 'the development and construction of new coastal and fluvial defences and possible marine landing jetties/docks could affect coastal processes, hydrodynamics and sediment transport processes at coastal and estuarine sites. These impacts could lead to localised or more widespread coastal erosion or accretion. There could also be changes to offshore features such as submerged banks and ridges and marine ecology.' Applicants will be expected to identify, and develop, appropriate mitigation measures to address the impacts on marine biodiversity and coastal geomorphology in their EIA.

Biodiversity and geological conservation

The potential impact of nuclear development on local ecology is well documented. The draft NPS sets out some common risks for biodiversity such as habitat/species loss and fragmentation, disturbance events (noise, light and visual) and air quality concerns. Applicants should develop an environmental management plan as part of their EIA. This will be an important aspect of demonstrating to the IPC that mitigation has been adequately factored into the application.

17.5 Key controls on environmental impacts

The environmental radiation risks associated with nuclear power generation have received a considerable amount of attention, and this is to some extent understandable given the occurrence of high-profile incidents such as Chernobyl and the damage caused by the tsunami at the Fukushima plant in Japan. The strong public opposition to nuclear power is often founded on environmental concerns associated with these rare occurrences, which overlooks the day-to-day threat posed in the construction, operation and decommissioning of plant. From water abstraction and discharges to industrial emissions and contamination to land, there are a host of different environmental impacts which will need to be tightly controlled. This is an area where the boundaries of regulation can converge, and which, in turn, requires the coordination of the activities of the authorities involved. The pervasive nature of environmental regulation in the lifecycle of a nuclear power plant (including at the design and planning stages) should not be understated, and it is therefore a subject which requires further consideration.

Environmental regulators

The body that is responsible for regulating environmental impacts, and its interaction with the body with overall responsibility for nuclear safety, is an important element of a developed nuclear institutional framework. The bodies charged with primary responsibility for regulating environmental impacts in the UK are the Environment Agency (EA) in England and Wales and the Scottish Environment Protection Agency in Scotland. Under the Environmental Act 1995 (their general duties and functions are set out in Sections 4–8), the EA is vested with primary responsibility, in relation to specified legislation (see Section 5(5)), including the Radioactive Substances Act 1993 (RSA 1993), Water Resources Act 1991 (WRA 1991) and the Environmental Permitting (England and Wales) Regulations 2010 (the 'Permitting Regulations 2010'), to use their powers 'for the purpose of preventing or minimising, or remedying or mitigating the effects of, pollution of the environment' (Section 5(1)). The operators of nuclear power plants will require a variety of different environmental permits from the EA, and the EA will monitor their activities (and impose reporting requirements) to ensure that they are complying with their permit conditions. The legislation empowers the EA to ensure compliance by providing them with the power to bring enforcement actions (see Part 4 of the Permitting Regulations 2010) against non-compliant operators, and to bring a halt to site operations by the revocation of permits where there have been serious or serial breaches (see Annex 1 of the EA's 'Submission to DTI – Pre-Licensing Assessments of New Nuclear Power Stations' for a general overview of the EA's regulatory responsibilities in relation to nuclear power).

Environmental regulation

Discharges and impacts to water

The environmental impacts caused by discharges, abstraction and the cooling water requirements of nuclear plant will require tight control by the environmental authorities. In the UK, operators will need to comply with the requirements of the WRA 1991. Part 2 of the WRA 1991 establishes a licensing regime for the abstraction and impounding of water and requires operators to apply for, and comply with, the terms of the licence, or face the consequences of a potentially unlimited fine. Part 3 enables the EA to control the pollution of water sources, and Section 84 imposes general duties on them to achieve and maintain water quality objectives established by the Department for Environment, Food and Rural Affairs. The core offence for water pollution is set out under the Environmental Permitting (England and Wales) Regulations 2010, and an operator will be in breach if he causes or knowingly permits any poisonous, noxious or polluting matter or any waste matter or trade effluent to enter any controlled waters otherwise than in accordance with an environmental permit. Operators will need to ensure that they have the requisite authorisations in place for their operations and that they have designed their environmental management systems so that they can monitor compliance with their discharge obligations. Sections 161 to 161D of the WRA 1991 give the EA powers to take action (including the power to issue works notices) to prevent or remedy the pollution of controlled waters. The water pollution offences are based on principles of strict liability (subject to the defence in Seciton 40 of the Environmental Permitting Regulations 2010 for discharges made in emergencies in order to avoid danger to life or health), which means that liability can be established in the absence of fault on the part of the operator. Liability has even been imposed where the acts leading to the contravention have been caused by third parties (e.g. vandalism) or natural events (see *Empress Car Co. (Abertillery) Ltd v National Rivers Authority* [1998]). As with a breach of the abstraction and impounding regime, there is no limit on the level of fine which may be imposed.

Discharges and impacts to air

Even though the environmental impacts to the atmosphere will be relatively minor in comparison to fossil fuel-based power generation, the atmospheric impact caused by incidental plant such as auxiliary boilers (e.g. for steam generation), back-up power generation facilities and incinerators used to dispose of combustible radioactive waste will need to be tightly controlled. In the UK, these installations are regulated by the EA under the Permitting Regulations 2010. These Regulations transpose the requirements imposed on EU Member States in the Integrated Pollution Prevention and Control (IPPC) Directive (Directive 2008/1/EC). The EA is required to exercise its functions 'for the purpose of achieving a high level of protection for the environment taken as a whole by, in particular, preventing or, where that is not practicable, reducing emissions into the air' (paragraph 3 of Schedules 7 and 8). Operators must ensure that their application for an environmental permit contains the information specified in Article 6(1) of the IPPC Directive which includes a description of:

(a) the sources of emissions from the installation;
(b) the nature and quantities of foreseeable emissions from the installation into each medium as well as identification of significant effects of the emissions on the environment;
(c) the proposed technology and other techniques for preventing or, where this is not possible, reducing emissions from the installation;
(d) measures planned to monitor emissions into the environment; and
(e) the main alternatives, if any, studied by the applicant in outline.

The EA will expect operators to design their plant in accordance with best available techniques and, as with other environmental permits under the Permitting Regulations 2010, will impose conditions that limit or control environmental impacts by requiring the achievement of specified environmental outcomes. Failure to achieve these outcomes may result in enforcement action under Part 4. For more serious breaches, the EA has the power to suspend and revoke permits, and to bring enforcement proceedings against companies and their negligent officers (which includes the possibility of up to two years' imprisonment).

Discharges and impacts to land: radioactive contaminated land

The contaminated land regime, introduced by Part 2A of the Environmental Protection Act 1990 (EPA 1990; see also the Contaminated Land Regulations 2006), was extended (with modification) to cover radioactive contaminated land by the Radioactive Contaminated Land (Enabling Powers) Regulations 2005 and the Radioactive Contaminated Land (Modification of Enactments) Regulations 2006 and 2007. In combination, they provide a regulatory system for identifying and requiring the remediation of contaminated land adjacent to nuclear sites where such land is causing lasting radiation exposure to any person, or where there is a significant possibility of such exposure. Extending the regime to cover radioactivity was necessary to ensure that the UK complied with its obligations to transpose Articles 48 and 53 of the Basic Safety Standards Directive (Directive 96/29 Euratom). The identification of contaminated land is based upon establishing a pollution linkage from a contaminant, through a pathway to a receptor. In the case of radioactive contaminated land, the receptor vulnerable to harm must be a human. The regime is based on the 'polluter pays' principle, unless the polluter cannot be found, in which case liability for remediation will shift to the owner or occupier of land (Section 78F of the EPA 1990). Determining responsibility may become an important issue where historic nuclear sites are being used for the commissioning of new plant.

Although the primary regulatory responsibility for contaminated land under Part 2A rests with local authorities, the EA is the enforcing authority where land is characterised as a 'special site' by virtue of radioactivity. A risk-based approach to remediation will be applied having regard to the anticipated costs and the seriousness of the risks or harm, and the enforcing authority is obliged to consider the contaminated land regime guidance document issued by the Secretary of State. The potential costs of contaminated land remediation could be extremely high, and a failure to comply with the terms of a remediation notice may result in the imposition of a fine, including the possibility of a daily fine for continued non-compliance (Section 78M). The enforcing authorities are also empowered to carry out the remediation themselves (Section 78BN) and to recover their reasonable costs from the party deemed responsible (Section 78P).

Waste licensing

Any disposal of non-radioactive waste (including excavation materials arising from construction) on a nuclear site will require an environmental permit issued by the EA under the Permitting Regulations 2010 and Part 2 of the EPA 1990 (see Section 33(1)(a)). Operators are prohibited from treating, keeping or disposing of controlled waste or extractive waste in a manner likely to cause pollution of the environmental or harm to human health. A duty of care is imposed on those in possession of waste requiring them to take all reasonable measures to, among other things, prevent the escape of waste from their control and secure that waste is only transferred to authorised persons. Like the other environmental permits issued under the Permitting Regulations 2010, the EA will be able to control the waste disposals through the conditions imposed, and to take enforcement action under Part 4.

Radioactive waste disposals

Recognising the unique and hazardous characteristics of nuclear waste, the UK operates a separate regulatory regime which is much more stringent than in other areas of environmental regulation. The accumulation and disposal of radioactive waste requires an authorisation granted by the EA under the Radioactive Substances Act 1993 (RSA 1993). A disposal includes those directly into the environment, for example discharges to air, water and land, as well as transfers to other sites for disposal (which includes treatment). The EA is obliged to consult with a number of bodies before granting an authorisation, including the HSE and 'such local authorities, relevant water bodies or other public or local authorities as appear . . . to be proper to be consulted' (Section 16(5); see also Section 18). Under the RSA 1993, the Secretary of State has retained key powers and can direct the EA to grant (with or without conditions), refuse, vary, cancel or revoke applications, and can require certain applications to be determined by him or her. The RSA 1993 regime offers different levels of regulatory control, from local authorities through to the Secretary of State, to ensure that environmental impacts are adequately reflected in radioactive waste management decisions. In order to ensure compliance, the EA has the power to issue enforcement and prohibition notices, and operators in breach of their authorisations could face an unlimited fine. There is also the possibility of up to five years' imprisonment where it can be proved that an offence has been committed, with the consent, or by the neglect, of an officer of a corporate body.

17.6 Overlap with other regulatory controls

An effective regulatory system will need to be able to coordinate the activities of the different authorities. It will be important to clearly demarcate their competences in order to reduce areas of overlap, and to design systems which encourage cooperation between regulators when necessary. In the UK, there is a split between health and safety controls, which are administered through site licences regulated by the HSE (ND) under the Nuclear Installations Act 1965 and the Ionising Radiations Regulations 1999, and environmental controls, which are predominantly overseen by the EA. In order to 'avoid duplication of effort and potentially conflicting demands as between radioactive substances regulations and those matters for which HSE is responsible' (Tromans, 2010, p. 295), the EA and the HSE entered into a Memorandum of Understanding in 2002 with the objectives (paragraph 6) of facilitating effective and consistent regulation by ensuring that:

(i) activities of EA and HSE in relation to nuclear licensed sites are consistent, coordinated and comprehensive;

(ii) the possibility of conflicting requirements being placed on licensees, or others operating on nuclear sites . . . is avoided;
(iii) synergies are exploited and the appropriate balance of precautions is attained;
(iv) duplication of activity is minimised; and
(v) public confidence in the regulatory system is maintained.

The Schedule to the Memorandum sets out the joint working arrangements between the EA and the HSE, in an effort to provide clarity to both the regulators and the regulated. There are various other interfaces of regulatory control (e.g. safety and security, transport) which will need to be carefully considered in the context of nuclear plant, and a similar level of coordination will be required.

17.7 Conclusions

As the European Commission has recently identified, one of the objections to SEA and EIA is that their benefits cannot be easily measured in financial or monetary terms. Nonetheless, there are a great number of benefits that arise from the SEA and EIA, and the benefits of carrying out the processes should be seen to outweigh the financial implications of preparing the assessment documentation. EIA and SEA bring benefits to any regulatory system which aims to establish harmonisation between the planning process and environmental integrity. This is fundamental in the context of nuclear energy as these considerations will be crucial in ensuring that energy policy is met with a degree of public approval. Not only do SEA and EIA ensure that environmental considerations are taken into account as early as possible in the decision-making process, they are ensure 'more transparency in environmental decision-making and, consequently, social acceptance' (European Commission, 2009, paragraph 2.4).

It will be equally important to consider how the planning and regulatory processes can be used to assess and control environmental impacts. Environmental considerations play a pervasive role in both processes, and will need to be carefully controlled. Planning systems should be designed so that the environmental risks can be sufficiently scrutinised. States will need to develop legal systems which adequately reflect the intended balance of powers, and which provide the government with residual influence and control over certain key decisions. The regulatory system will need to be reinforced by a sanctioning regime which is stringent enough to compel compliance. Although the UK approach is instructive, other jurisdictions will ultimately need to determine for themselves what role the environment will play in shaping the future of nuclear power generation. The various examples of other established civil nuclear states, multinational regulations and practices, international conventions and IAEA standards are also valuable sources for emerging civil nuclear states.

17.8 Future trends

In April 2010, the European Union Committee of the Regions published an Opinion entitled 'Improving the EIA and SEA Directives' (European Union Committee of the Regions, 2010). Besides affirming the importance of the SEA Directive and EIA Directive as tools in environmental protection, the Committee recognised that certain gaps remain in ensuring that the processes realise their objectives. Perhaps unsurprisingly, one of the main proposals of the Committee is that the EIA Directive should be amended so as to incorporate thresholds, criteria or triggers for the purposes of determining the significance of environmental impacts caused by Annex II activities. The Committee highlights the fact that certain Member States, when implementing the EIA Directive, have been shown to exceed their powers of discretion by only taking account of certain Annex III selection criteria or by completely exempting certain types of project in advance. The Committee also makes recommendations that the assessment of alternative solutions should be made obligatory, and that gaps in public participation procedures should be addressed by giving the public early and effective opportunities to participate from the earliest possible point. The Committee makes few concrete recommendations in relation to the SEA Directive, principally by virtue of the fact that further experience in applying the SEA Directive is required. However, certain issues are identified, notably that a specific definition of reasonable alternatives on a mandatory basis should be developed, that it should be made obligatory to establish methods and indicators of monitoring environmental impacts, and that the SEA Directive should better identify what information the Environmental Report should contain.

The coalition government in the UK has decided to abolish the IPC and replace it with the MPIU, but decisions will still be taken in accordance with NPSs. The UK remains committed to new nuclear power despite the serious events which occurred at the Fukushima nuclear plant in Japan. With a number of major new build planning applications in the pipeline, it will be interesting to assess how the UK balances environmental impacts with other factors such as energy security and climate change. There will no doubt be considerable environmental opposition, and any favourable decisions will be potentially subject to legal challenge.

17.9 References

Aarhus Convention on Access to Information (1998), Public Participation in Decision-Making and Access to Justice in Environmental Matters, Aarhus, Denmark

Berkeley v Secretary of State for the Environment and Another [2001] 2 A.C. 603
Case C-205/08 *Umweltanwalt von Kaernten*
Case C-431/92 *Commission v Germany, paragraph 39*
Contaminated Land Regulations 2006 SI 2006/1380
Department for Business, Enterprise & Regulatory Reform (2008), *Meeting the Energy Challenge: A White Paper on Nuclear Power*, London, BERR
Department for Energy and Climate Change (2009), *Draft National Policy Statement for Nuclear Power Generation (EN-6)*, paragraphs 3.4.4 and 4.4.1, p. 27
Department for Energy and Climate Change (2009), *Draft Overarching National Policy Statement for Energy (EN-1)*
Department of Trade and Industry (2006), *The Energy Challenge: Energy Review Report*
Directive 2001/42/EC of the European Parliament and of the Council of 27 June 2001 on the assessment of the effects of certain plans and programmes on the environment
Directive 2008/1/EC of the European Parliament and of the Council concerning integrated pollution prevention and control
Directive 96/29/Euratom laying down basic safety standards for the protection of the health of workers and the general public against the dangers arising from ionizing radiation
Electricity Act 1989
Empress Car Co. (Abertillery) Ltd v National Rivers Authority [1998] 1 All ER 481
Environment Act 1995
Environment Agency (*c.* 2006), The Environment Agency's Submission to DTI – Pre-Licensing Assessments of New Nuclear Power Stations and Streamlining the Regulatory Process
Environmental Assessment of Plans and Programmes Regulations 2004 SI 2004/1633
Environmental Permitting (England and Wales) Regulations 2010 SI 2010/675
Environmental Protection Act 1990
EU Council Directive 85/337/EEC of 27 June 1985 on the assessment of the effects of certain public and private projects on the environment
European Commission (2009), Report of the Commission to the Council, the European Parliament, the European Economic and Social Committee and the Committee of the Regions Commission, *On the application and effectiveness of the EIA Directive 85/337/EEC, as amended by Directives 97/11/EC and 2003/35/EC,* COM (2009) 378 final, European Commission, Brussels, paragraph 3.5
European Convention on Human Rights and Fundamental Freedoms (1950)
European Union Committee of the Regions (2010), Opinion on *Improving the EIA and SEA Directives*, 15 April 2010, CdR 38/2010
Infrastructure (Environmental Impact Assessment) Regulations 2009 SI 2009/2263
Infrastructure Planning (Applications: Prescribed Forms and Procedure) Regulations 2009 SI 2009/2264
International Atomic Energy Agency (2006), *Fundamental Safety Principles (no. SF-1)*
International Atomic Energy Agency (2007), *Milestones in the Development of a National Infrastructure for Nuclear Power (no. NG-G-3.1)*
International Atomic Energy Agency, International Nuclear Safety Group (2006) *Stakeholder Involvement in Nuclear Issues (INSAG-20)*

International Commission on Radiological Protection, 1990 Recommendations of the International Commission on Radiological Protection, 60, *Annals to the ICRP*, Volume 21, Issues 1–3, Pergamon Press, Oxford, pp. 1–201

Ionising Radiations Regulations 1999 SI 1999/3232

Justification of Practices Involving Ionising Radiation Regulations SI 2004/1769

Kelly B (2008), The Planning Bill: Implications of the proposals for a new regime for major infrastructure for democracy and delivery, *Journal of Planning and Environment Law*, 13, supplement, p. 2

McCracken R (2009), Infrastructure Planning Commission: challenge or opportunity? *Journal of Planning and Environment Law*, 13, pp. 7–23

Memorandum of Understanding between the Health and Safety Executive and Environment Agency on Matters of Mutual Concern at Nuclear Sites Licensed by HSE in England and Wales (23 April 2002)

Nuclear Installations Act 1965

Office for Nuclear Development (2009), *Towards a Nuclear National Policy Statement*

Office of the Deputy Prime Minister (2005), *Planning Obligations (Circular 05/2005)*

Planning Act 2008

R (Greenpeace Ltd) v Secretary of State [2007] EWHC 311 (Admin)

Radioactive Contaminated Land (Enabling Powers) Regulations 2005

Radioactive Contaminated Land (Modification of Enactments) (England) Regulations 2006 SI 2006/1379

Radioactive Contaminated Land (Modification of Enactments) (England) (Amendment) Regulations 2007 SI 2007/3245

Radioactive Substances Act 1993

Stoiber C, Baer A, Pelzer N and Tonhauser W (2003), *Handbook on Nuclear Law*, Vienna, IAEA, pp. 34–35

Town and Country Planning Act 1990

Treaty Establishing the European Atomic Energy Community (1992, Maastricht)

Tromans S (2010), *Nuclear Law: The Law Applying to Nuclear Installations and Radioactive Substances in its Historic Context*, London, Hart Publishing, pp. 141, 145, 295

United Nations Convention (1991), Convention on Environmental Impact Assessment in a Transboundary Context, adopted 25 February 1991, United Nations, New York

United Nations Environment Programme (16 January 1987), *Goals and Principles of Environmental Impact Assessment*, United Nations, New York

United Nations Environment Programme (1992), *Rio Declaration on Environment and Development*, Principle 17, United Nations, New York

United States Regulatory Commission, *Environmental Protection Regulations for Domestic Licensing and Related Regulatory Functions (10 CFR Part 51)*

US National Environmental Policy Act 1969

Water Resources Act 1991

18
Site selection and evaluation for nuclear power plants (NPPs)

A. ALONSO, Universidad Politécnica de Madrid, Spain

Abstract: This chapter describes the technical requirements to be considered in the selection of a site for a nuclear power plant. The design and operation of the nuclear power plant depend on the site characteristics; the site-derived risks have to be considered in the plant design basis, and the site itself has to bear the risks and detriments coming from the plant. The design has to cope with expected extreme natural phenomena and combinations of those, as well as human-induced events, without impairing the operational safety of the plant. The site has to provide needed requirements such as rejected and decay heat sinks, availability of electrical power supplies, good communications and effective emergency management, including the evacuation of nearby residents.

Key words: site evaluation, site seismicity, extreme meteorology, ultimate heat sink, population density, site parameters, human-induced events.

18.1 Introduction

Site selection for nuclear power plants (NPP) requires the analysis of a set of diverse parameters which are divided into the following five groups:

- Conventional industrial factors
- Site-related hazards that determine the safety of the plant
- Physical circumstances that determine the radiological impact of the plant under normal operation and accident conditions
- Parameters that determine the physical impacts from the plant into the site
- Conditions that determine the social and economic impacts from the plant on the local population.

Among the conventional industrial factors the economic, local and technological related ones are relevant. Economic factors require one to prove the need for the plant, the convenience of the selected site and the proximity of an electricity market. Local parameters such as accessibility, heavy equipment transportability and availability of human resources, mainly during the construction phase, become significant. Technological parameters demand the accessibility to an ultimate heat sink, the atmosphere or a large water body, a redundant electrical net and availability of construction materials. These conventional factors, although relevant, for NPP site selection are not developed further. Some of these elements are considered in the chapter on site and supporting facilities in the IAEA 'Milestones' document (IAEA, 2007a).

Site characteristics which may impact plant safety are essential in the safe design of the NPP. Extreme meteorological conditions, such as hurricanes, tornadoes and heavy rain, hail or snow falls and lightning may produce floods and impair plant accessibility; floods can also be produced by large tides in combination with heavy rain in estuarine waters or by rupture of up-river dams; earthquakes and the ensuing tsunamis may produce damage to buildings and external facilities as well as violent floods in coastal sites. All these events require buildings, water intake structures, external water tanks, electrical grids and electrical transformers to be protected by design. These aspects will be considered in depth.

Man-made external activities offer risks to the safety of the plant. Large explosions and toxic releases in the vicinity of the plant produced, for example, in the transportation of liquefied natural gas or other explosives or gaseous toxic substances by road, rail or waterways have to be avoided and the consequences to the plant mitigated by design. Proximity to harbours, airports and military installations should also be avoided as they represent a risk to the safety of the NPP. These hazards are also considered in detail.

Extreme natural events, such as the impact of large meteorites, massive volcanic activity or pandemics are not generally considered. There should also be protection against sabotage, but this matter is not considered in this chapter, as it is not site dependent and it is generally prevented by security measures.

The NPP creates risks and detriments to the surrounding population and the environment that have to be previously analysed. The impact of radioactive releases during normal operation requires meteorological and hydrological dispersion parameters. The transport of contaminants along the terrestrial and aquatic food chain pathways needs to be considered. Demographic parameters are needed to assess the potential doses that different population groups may receive and the performance of epidemiological studies to determine any potential radiation effects. These risks and detriments are considered.

A major consideration is the protection of the population in case of accidents with radiological effects. The site conditions should facilitate the evacuation of the affected population, the establishment of decontamination and receiving centres and the medical treatment of potential exposures. Population distribution, communication and evacuation routes are relevant aspects of site evaluation. The site requirements for an effective nuclear emergency plan are emphasized.

Most countries have regulated that any major industry or activity has to analyse the environmental impact it may produce. A major environmental impact of the NPP is due to the rejection of heat required by the second law of thermodynamics. The thermal efficiency of NPPs is about one-third, therefore two-thirds of the heat generated has to be rejected to an ultimate heat sink, which could be the atmosphere through different types of cooling towers, or large water bodies such as big rivers, natural or artificial lakes, or the sea in coastal sites. Cooling towers release large amounts of steam to the atmosphere with minor meteorological impacts on the surroundings, and heat rejected to water bodies, causing even small temperature increases, may produce substantial variations in the life and development of aquatic organisms. NPPs also produce chemical impacts, mainly in water bodies; production and releases of conventional waste; an increase in light and heavy traffic, mainly during construction; and a substantial aesthetic impact, mainly if using large cooling towers. These impacts are not considered in detail in this chapter. Such aspects are described in Chapter 8.

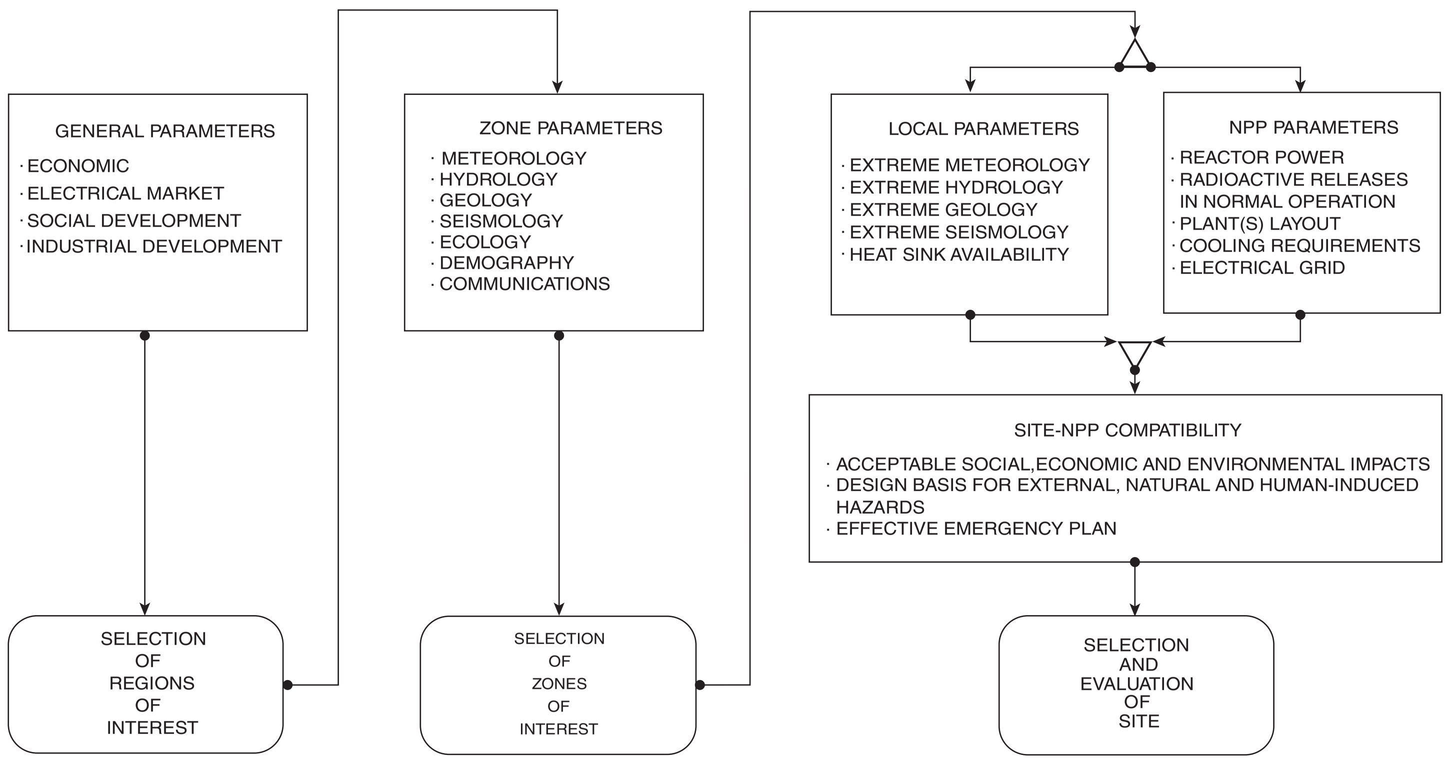
All the effects named above require specific studies and consideration. In this chapter safety parameters will receive more attention. The basic scientific basis will be exposed and references to applicable International Atomic Energy Agency (IAEA) standards will be introduced to serve as additional information.

18.2 Schematic approach to site selection

The selection of any new site for a large industrial installation requires a systematic approach, as described in Fig. 18.1. First of all, a whole country or a region of it is selected based on economic considerations, proximity to an electricity market and social demands, such as the convenience of boostering the development of a particular region.

Within the region, selection of one or more zones will have to be determined by a general analysis of some basic parameters. At this first stage, the availability of cooling water is the most restrictive technological requirement; generally the areas of interest are limited to rivers, lakes or coastal sites. Artificial lakes could also be built on smaller tributaries and the use of cooling towers may open more possibilities, although the proximity to a large body of water is always recommended.

An analysis of the geology of the region originally selected will determine which areas must be discarded because of high seismicity or for other reasons; the meteorology of the region will determine the hazards associated with extreme meteorological events; zones that are densely populated or near large population centres (over 25,000 residents) will also be discarded as it would be difficult to establish efficient emergency procedures; and sites close to large industrialized areas or areas with high agricultural



18.1 NPP selection process.

or ecological value should also be dismissed. The systematic approach of all these criteria will divide the region in question into zones or areas in which NPPs may be situated. At this time, a gradation of the areas found could also be established. Countries have evaluated the maximum nuclear capacity which could be installed along a given river, large lake or coastal region.

The zone or zones of interest are then studied in depth to determine the optimum sites where the plant or plants could be built. Two types of studies are conducted; on the one hand, detailed economic, geological, hydrological, meteorological and social and demographic studies will determine some basic plant design parameters, while on the other hand the plant or plants to be constructed will determine some basic requirements from the site, mainly related to the ultimate heat sink, redundant electrical power supply, the need to have an efficient emergency management system, the release of radionuclides during normal operation and accident conditions, the management of radioactive waste, and the size and geometries of the buildings to be erected.

The initially separated studies described above are later subjected to a compatibility study between the plant and the site, covering all types of parameters which constitute the basis for the selection. The site information gathered and the compatibility of the site and the selected technology or technologies will constitute the basis for requesting the site approval from the Regulatory Body in accordance with the regulations of the country.

In the past, governmental institutions and large utilities, under their areas of influence, have conducted studies to determine the best locations for building nuclear power plants and fuel cycle installations. For new entrants the development of such a bank of potential sites is highly recommended. The current social opposition to nuclear power makes it difficult to find new sites for nuclear power plants and related facilities.

18.3 Basic safety principles applicable to nuclear power plant (NPP) siting

INSAG has established four specific safety principles applicable to the siting of a NPP. They address the following issues: the external factors affecting the plant; the radiological impact on the public and the local environment; the feasibility of emergency plans; and the ultimate heat sink provision (INSAG, 1999).

The principle for the external factors affecting the plant is formulated as follows:

> *Principle 1: The choice of site takes into account the results of the investigation of local factors that could adversely affect the safety of the plant.*

This principle recommends the identification of local factors which must be considered in the design of the plant. They are generally classified into two groups: natural events and those created by human activities. Among the first, the characterization of seismic events and geological, hydrological and meteorological extreme disturbances are the most relevant. Among the second, contaminations, explosions and deflagrations of flammable and toxic gas releases in the proximity of the plant are the major concerns. The studies are aimed at evaluating the expected frequency of these natural phenomena and human-induced acts as a function of their magnitude. The designers need the magnitudes and characteristic parameters of all these natural and human-induced events to be sure that they will be properly included in the design basis in such a way that the plant will cope with the phenomena under consideration.

The principle concerning the plant's radiological impact on the public and the local environment is presented as follows:

> *Principle 2: Sites are investigated from the standpoint of the radiological impact of the plant in normal operation and in accident conditions.*

The analysis of the radiological impact on the surrounding population and the environment requires the analysis of the vectors causing the dispersion of radioactive nuclides, i.e. the wind and the water, the use of the land and water bodies, the food chain pathways, population distribution and habits. All these studies will serve to limit the radioactive releases to air and water in such a way that the safety objectives are fulfilled in normal operation and countermeasures introduced into the design basis to limit the consequences from accidental releases of radioactive nuclides.

The principle on the feasibility of the emergency plans is offered in the following way:

> *Principle 3: The site selected for a nuclear power plant is compatible with the off-site countermeasures that may be necessary to limit the effects of accidental releases of radioactive substances, and is expected to remain compatible with such measures.*

The emergency plan is considered the last barrier available to protect people against the harmful effects of radiation coming from the liberated radionuclides. It demands a substantial national administrative and technical infrastructure which is reflected in the corresponding emergency plan. The site characteristics require that there should be an efficient way of communicating the situation to the affected people. Among the emergency procedures, in order of increasing importance, it will be necessary to remain indoors, ingest potassium iodide to protect the thyroid, and evacuate people to safer places. Among the long-term measures it is necessary to monitor water and food, to confiscate crops and other products and to establish a decontamination programme. Large population densities, intensive industrial and agricultural development, complicated topography and lack of evacuation routes are impediments to an efficient emergency plan.

The principle on the ultimate heat sink provisions is defined as given below:

> *Principle 4: The site selected for a nuclear power plant has a reliable long-term sink that can remove energy generated in the plant after shutdown, both immediately after shutdown and over the longer term.*

The generation of residual energy after reactor shutdown due to the disintegration of the radioactive fission and activation products, the so-called decay heat, is a specific property of nuclear power. The impossibility of removing such energy causes the heating up of the core, the loss of fuel integrity and its potential meltdown and the release of radionuclides. Therefore the availability of an ultimate heat sink is an unavoidable requirement. This ultimate heat sink could be the same as the sink receiving the heat rejected from the thermodynamic circuit, but under accidental conditions the decay heat can be released to the atmosphere providing that such systems will withstand all foreseeable extreme circumstances.

18.4 International Atomic Energy Agency (IAEA) requirements and safety guides on nuclear power plant siting

The IAEA has provided a complete and satisfactory set of requirements and safety guides on NPP siting through its safety standards series. The safety requirements document (IAEA, 2003a) includes a list of general requirements, a list of specific site requirements for evaluating the effects of external events on the plant safety, and the potential effects of the plant on the site and its surroundings.

18.4.1 General requirements

General requirements are based on the four principles described in Section 18.2. The principle concerning external factors affecting the plant obliges the applicant to investigate all possible natural phenomena and human-induced situations which may constitute a hazard to safe operation of the future NPP. For natural phenomena such studies require the analyses of prehistoric, historical and currently instrumented information and records related to the phenomena under study. For human-induced situations, they are of use in evaluating the hazards associated with hazardous industries and activities, such as the transport of explosive, flammable and toxic materials, around the site under consideration and their foreseeable development with time.

The principle concerning the radiological impact on the public and the local environment requires the investigation of all potential radiological impacts on people and the environment that may be produced from the expected release of radioactive nuclides during normal operation and accident conditions.

The principle concerning the feasibility of emergency plans requires the evaluation of the present and future distribution of the population in the region of interest, the present and foreseeable future uses of land and water and the radiological risks that the affected population may support in case of accident. It is also necessary to evaluate which site characteristics or concurrent natural phenomena may occur that may hinder the efficiency of the already established emergency plan. The Fukushima event has clearly demonstrated how the major emergency situation created by the earthquake and resulting tsunami created a nuclear emergency inside a previous, much larger, naturally induced emergency.

The principle related to the ultimate heat sink provision requires the identification of the pathways internal to the plant by which decay heat can be transferred to the environment under heavily deteriorated conditions. Damping such decay heat to the atmosphere is recommended provided that such releases do not involve the concurrent release of radioactive nuclides.

18.4.2 Specific site requirements for external events

The IAEA requirements document divides external events into six major groups:

- Earthquakes and surface faulting
- Meteorological events
- Flooding
- Geotechnical hazards
- External human-induced events
- Other important considerations.

Earthquakes and surface faulting

The requirements clearly indicate that 'the hazards associated with earthquakes shall be determined by means of seismotectonic evaluation of the region with the use of the greatest possible extent of the information collected', while the selected site requires an analysis of the fault capability existing there.

The earthquake and surface faulting requirements have been extended into several safety guides; there is a dedicated safety guide on seismic hazards on site evaluation (IAEA, 2010), where details on how a regional seismotectonic model and local surface faulting can be developed and how to quantify the seismic hazard by using deterministic and probabilistic approaches. These data serve to define the operating basis earthquake (OBE) and the safe shutdown earthquake (SSE) which constitute the basis of the plant's seismic design.

Meteorological events

The requirements divide the meteorological events into two major groups: extreme meteorological phenomena and rare meteorological events. The extreme values of phenomena such as wind velocity, precipitation, snow packs, temperatures, seawater levels and storm surges should be measured to determine the design criteria for the affected structures and components. When using the probabilistic approach the probability of such values being exceeded should be given together with the associated uncertainties.

Rare meteorological events include lightning, tornadoes and tropical cyclones. The hazards associated with the above phenomena should include the expected frequency of occurrence and the expected maximum values of each phenomenon's defining parameters: maximum rotational wind speed, pressure differences and rate of change of pressure for tornadoes, and wind speed pressure and precipitation for tropical cyclones. In both cases, missiles with potential to harm the plant generated by the phenomenon itself should also be contemplated.

An IAEA safety guide on meteorological events in site evaluation (IAEA, 2003b) describes how data should be collected from extreme meteorological phenomena and rare meteorological events, how to derive the hazards associated with such phenomena and events, and how to obtain the values needed for the design of structures and components potentially affected.

Flooding

The requirements divide flooding into three major causes: floods due to precipitation and other causes; water waves induced by earthquakes or other geological phenomena; and floods and waves caused by failure of water control structures. Flooding can be caused by one or more concurrent natural phenomena, such as heavy precipitation and rapid snow melt, high tides and storm surges, seiche and wind waves, among many other combinations. Flooding hazards from tsunamis associated with marine earthquakes or close to lakes and large rivers, and seiches originating from geological causes should be quantified to design protective measures. On river sites, floods and waves caused by the failure of upstream dams or other water retention structures have to be analysed to determine the level above the river surface where safety structures should be built to avoid floods or to protect key structures, systems and components.

An IAEA safety guide considers flood hazards for nuclear power plants sited on coastal and river sites (IAEA, 2003c). The guide considers each one of the flooding causes and identifies deterministic and probabilistic parameters which should be included in the design basis of the plant. Due consideration is given to the concurrence of different causes of flooding and the potential changes in the initial parameters due to climate changes or geographical modifications. The stability of shorelines and river beds is also a matter for consideration.

Geotechnical hazards

Geotechnical hazards are caused by slope instability, collapse, subsidence or uplift of the site surface, solid liquefaction and anomalous behaviour of foundation materials. Slope instabilities can be produced by landslides or snow avalanches with the possibility of affecting the electrical grid and the water intakes; caves, karstic formations, underground rivers, mines and water, gas or oil wells may cause collapse, subsidence or even uplifts of the site surface, affecting building structures; solid liquefaction induced by earthquakes has the potential of causing differential movements among builds connected by pipes and cables; and foundation materials may also produce differential movements among buildings, requiring a good knowledge of the foundation materials and their properties under static and seismic loadings.

An IAEA safety guide addresses the geotechnical aspects to be considered in site evaluation and foundations for NPPs (IAEA, 2005). The guide describes the many geological aspects that must be evaluated, the observations to be conducted and the laboratory tests to be performed. Special attention is given to the stability of the site and the foundations, necessary for the design basis of the major buildings. Such parameters, mainly the stability of the foundations, need to be reassessed during site preconstruction activities and monitored during the operational life of the plant.

External human-induced events

The requirements consider three types of external events: aircraft crashes, chemical explosions, and other important human-induced events. The requirements consider only accidental events of that type; terrorist attacks by the same means with the only purpose of producing harm are excluded.

The currently high level of air traffic and the expected increase in the future may represent a substantial hazard to NPPs, despite the high level of safety in current and future air flights. The hazards due to the crash of a large passenger airplane include the impact itself, the explosions and large fires that may be produced. The risk can be reduced by avoiding having flight corridors close to NPPs.

The proximity of chemical installations and transportation routes also creates risks of explosions, deflagrations and large fires with the corresponding release of toxic substances that may affect the safety of the operating NPP; compensatory measures should be applied or the site discarded. Other important human-induced events include large fires, for instance forest fires, collisions of ships with water intake structures, and the presence of electromagnetic waves with the potential of affecting the plant information, control and instrumentation systems.

An IAEA safety guide describes the many external human-induced events to be considered in the evaluation of a site for a NPP (IAEA, 2002a). The guide considers each one of the events, defines the associated hazards and determines how to obtain the main parameters to be used in the design basis of the plant to cope with such hazards.

Other important considerations

The requirements also consider a rather long list of local phenomena that may produce harm to the safety of the NPP, such as volcanism, sandstorms and subsurface freezing of sub-cooled water. There are also phenomena which may have an impact on the long-term removal of decay heat; in this respect consideration should be given to air temperature and humidity, water temperature, available flow of water and natural and human-induced phenomena which could impair the loss of the heat removal function, such as insufficient river flow, loss of the available water reservoir, water intake blockage by marine organisms or freezing of cooling towers, among others, All these aspects have to be considered to include appropriate preventative and mitigation devices, equipment and procedures.

The IAEA has not yet developed any safety guide to measure the hazards associated with these varied concerns. Only a safety guide on volcanic hazards in site evaluation for nuclear installations is in preparation (IAEA, 2009). The guide, based on a previous 1997 document, includes the knowledge gained in the science of volcanology and associated risks mainly due to the enormous amount of volcanic ashes that are injected into the upper atmospheric levels, which is already evident in the effects produced on air traffic. The fallout of a large amount of ashes on the plant premises, water intakes and roads could impair the safe operation of a nuclear power plant. As in the past, there could also be mega-volcanic eruptions with the potential of blocking sunlight, which may have a serious impact on all types of installations and on life on earth. These extreme effects are generally outside the scope of the design.

18.4.3 Potential impacts of the nuclear power plant on the site and its surroundings

The potential impacts that an operating NPP may bring to the site are related to the atmospheric, surface and ground water dispersion of radioactive material affecting the population and the use of land and water in the affected region. Specific requirements and corresponding safety guides follow.

Atmospheric dispersion of radioactive materials

During NPP operation small amounts of radioactive nuclides are released to the atmosphere under strict control. Those nuclides include some noble gases which cannot be retained by any treatment process, as well as some volatile elements and particulate matter which may not be completely retained in the high-efficiency radioactive waste treatment system.

In pressurized water reactors (PWRs) fission and activation gases in the coolant are separated and stored for decay and finally vented to the atmosphere before refuelling outages; most of these gases have short lives and disappear during the storage period with the exception of krypton-85 (half-life 10.6 years) which becomes the larger contributor of gaseous releases. Iodine-131 (half-life 8.06 days) and hot particles – radioactive particles including fission and activation generated radioactive nuclides – could also be found in the containment atmosphere from coolant leakages. Small amounts of such materials can also be released to the atmosphere after being filtered by activated carbon filters, to retain iodine, and high-efficiency particle air filters. In boiling water reactors (BWRs) radioactive gases generated in the coolant are carried by the steam, separated in the condenser and released to the atmosphere continuously after passing for a few days through a delay system composed of a large activated carbon bed maintained at low temperature; short-life nuclides decay during transit through the bed with the exception of krypton-85 and xenon-137 (half-life 5.27 days), while most of the iodine nuclides are retained in the activated carbon bed.

The behaviour of the released radioactive nuclides has to be predicted to measure the potential effects of such releases on the health and safety of the affected people and on the environment. Meteorological dispersion parameters have to be measured by dedicated meteorological towers to determine wind speed and direction, air temperature, precipitation, humidity and atmospheric stability parameters. With all these data sophisticated dispersion models are developed which also take into account the topography of the place and the effects that buildings may have on the dispersion of the released materials. These studies are conducted at least one year before starting plant construction and meteorological towers and data gathering are maintained during the whole operating life of the NPP.

The availability of meteorological data at the time and during an accidental release of radioactive products is essential to emergency management. In these cases the site dispersion data have to be supplemented with the data and dispersion models of the country's central meteorological agency.

An IAEA safety guide has been developed to indicate the ways and means to obtain such data and develop the site dispersion models (IAEA, 2002b). The guide describes how to select and display a meteorological data gathering system appropriate to the topography and physical characteristics of the region where the plant is located. The guide also describes methods to develop an *ad hoc* dispersion models which will also include the contamination of soil and the potential resuspension of radioactive aerosols and hot particles, as well as contamination of vegetables and other food products. Such models are essential parts of the theoretical estimation of the potential radiation doses that the affected population may receive from atmospheric radioactive releases by direct exposure to the radioactive cloud and other pathways. The model is also essential in the epidemiological studies generally conducted around nuclear sites.

Dispersion of radioactive material through surface water

Surface water may get contaminated through direct discharges to water bodies or through fallout from radioactive clouds. All contaminated waste water from the operation of a NPP is collected and treated in the liquid waste treatment system. This normally includes a high efficiency filtration system to retain particulate materials, followed by ion exchange to retain dissolved ions and also by a residual water evaporation system. At the end of these treatments pure distilled water is obtained; it can be recycled into the plant but any surplus of water has to be released to a nearby water body under strict control. Although of high efficiency, all these waste water treatment processes cannot retain all radioactive materials; moreover tritium substitutes hydrogen in the water molecule and it cannot be separated by any of the three processes mentioned above.

Tritium is a fission product – about one tritium atom is produced every 10,000 fissions. In PWRs tritium is also produced by activation of boron added to the coolant to control reactor core reactivity that is used in such reactors and with lithium also used in PWRs to control water pH. Tritium (half-life 12.26 years) is continuously produced in the stratosphere by cosmic radiation reacting with oxygen and nitrogen, from the stratosphere it gradually descends to the lower parts of the atmosphere by natural diffusion, and it ends in the ocean and terrestrial waters. As it is generated and decays constantly, it has reached an equilibrium estimated at about one million curies. It is against this natural background that tritium generated in NPPs and reprocessing facilities has to be measured.

The behaviour of the released nuclides to surface water, especially tritium, has to be predicted to measure the potential effects of such releases on the health and safety of the affected people and on the environment. Surface water dispersion and dilution parameters have to be measured by dedicated processes to determine water flows, and the transfer mechanism by which nuclides may reach humans. With all these parameters dispersion models are developed which also take into account all potential phenomena that control the behaviour of the different contaminants. These studies are conducted at least one year before starting plant construction and are maintained during the whole operating life of the NPP.

As in the case of meteorological data, surface water dispersion and dilution data and models are essential at the time and during an accidental release of radioactive products to properly manage the emergency use of such waters. In these cases dispersion and dilution data have to be supplemented with the data and dispersion models of the country's central hydrological agency. The 2011 Fukushima event released substantial amounts of contaminated water with radioactive nuclides to the Pacific Ocean with the potential of becoming concentrated in the bodies of fish and marine food products, thus requiring the definition of accepted concentration limits for human consumption.

The IAEA safety guide already mentioned (IAEA, 2002b) indicates the ways and means to obtain surface water data and develop the site water body's dispersion models. The guide describes how to select and display a data gathering system appropriate to the hydrology of the region where the plant is located. Differences between river, open coastal, estuarine and artificial lake receivers are marked. The guide also describes methods to develop appropriate dispersion and dilution models to clearly determine the different pathways through which contamination can reach humans. Such models are also an essential part of the theoretical estimation of the potential radiation doses that the affected population may receive from surface water pathways. The model is also essential in the epidemiological studies generally conducted around nuclear sites.

Dispersion of radioactive material through ground water

Ground water may be contaminated by leakages from buried pipes carrying contaminated fluids, through seepage and infiltration of surface water that has been contaminated and from interactions with contaminated surface waters. Several instances of tritium presence in ground water from NPP underground leaking pipes have been recently reported. Ground water uses include human consumption and irrigation, pathways that may bring tritium into human contact. Therefore the protection of aquifers from such events should be prevented and a geological barrier should be considered.

A description of the ground water hydrology at the local and regional level is then required to assess the behaviour of any contaminant, its potential migration and dilution, the retention characteristics of the soil and the physicochemical properties of the materials, mainly its retention properties. From this information a model is developed to estimate the radionuclide pathways during normal operation and under accident conditions.

As in the case of surface waters, IAEA (2002b) describes which data should be collected, that may require drilling boreholes for geophysical and tracer studies. The models will serve to estimate the expected contamination of ground waters at the point of use, and to assess the doses received by the exposed population and for the management of such waters in case of accident.

18.4.4 Population distribution and local and regional uses of land and water

The safety objective is to protect individuals and society as a whole against the harmful effects of ionizing radiation during normal operation and also under accident conditions. To accomplish that goal, population inventory and population distribution is required, as well as the local and regional uses of land and water. The atmospheric, surface water and ground water radioactive contaminant pathway models, together with water and land uses, serve to identify the most exposed population groups, the so-called critical groups, as a reference for normal operation. Such models could also serve to establish an efficient emergency plan covering the whole affected population in case of accident.

As in previous cases, the IAEA site requirements establish that data should be obtained on existing and projected population distribution, both resident and transient; such data is updated along the operating time of the power plant, at least every 10 years. Likewise, the present and future uses of land and water must be determined and updated along the lifetime of the plant. The area of interest is country dependent. In general, large population centres should not exist within a radius of 5 km and population distribution should cover at least a radius of 10 km from the plant. Proximity to schools, hospitals and prisons should be avoided.

Details on how to collect and gather the required information can be obtained from the IAEA safety guide (IAEA, 2002b).

18.5 Consideration of the feasibility of an emergency plan

Principle 3 of the INSAG document on NPP siting described in Section 18.3 addresses the study of the feasibility of an emergency plan in the site selected. Emergency planning, the last barrier to protect the health and safety of the population, has a considerable relevance. In the IAEA safety guide already quoted (IAEA, 2002b) the site-related aspects of nuclear emergencies are introduced: 'There should be no adverse site conditions which could hinder the sheltering or evacuation of the population in the region or the ingress or egress of external services needed to deal with an emergency.' Sheltering in people's own houses is the most elementary way to protect people; to make sheltering effective some basic procedures have to be put in place. Electricity and sufficient water and food should be available; special population groups such as residents in hospitals and prisons will also demand special services.

Poorly developed transport and communications networks or the presence of industrial activities may impair the rapid and free movement of people and vehicles in case of evacuation to safer places. Such places should be defined and be prepared beforehand, with alternatives in case they also become contaminated. In case evacuation routes have to pass close to the affected plant new routes have to be open. The Chernobyl-4 and Fukushima-1 accidents have demonstrated the need for permanent or prolonged displacement, a situation that needs government attention. The cited IAEA safety guide includes the following list of items to be considered for an efficient emergency plan:

- Population density and distribution in the region
- Distance of the site from population centres
- Special groups of the population who are difficult to evacuate or shelter, such as people in hospitals or prisons, or nomadic groups
- Particular geographical features such as islands, mountains and rivers
- Characteristics of local transport and communications networks
- Industrial facilities which may entail potentially hazardous activities
- Agricultural activities that are sensitive to possible discharges of radionuclides
- Possible concurrent external events.

The last item has particular interest. Evacuation may have to be conducted under heavy fog or snowfall or concurrent with other major natural phenomena such as an earthquake and tsunami as in the case of the 2011 Fukushima event.

The IAEA has developed a series of requirements and safety guides on emergency planning. A requirements document (IAEA, 2002c) addresses the logistic support and facilities needed as well as the training drills and exercises which should be conducted on a periodic basis. These requirements are further developed in a safety guide (IAEA, 2007b) in which Appendix VIII describes the conditions that emergency facilities and locations should comply with.

18.6 Demographic requirements and site parameters developed and applied by the United States Nuclear Regulatory Commission

The relevance of site characteristics, mainly population distribution, on the safety of NPPs was soon recognized. The ideas and concepts developed by the United States former Atomic Energy Commission (AEC) since 1962 have been maintained until now and have provided the basis for evaluating the site for NPPs; they also have shaped the design basis of the currently operating reactors. Because of the large impact of these considerations, it has been found of interest to present a short account of this historical development and the current situation regarding this matter.

18.6.1 A short historical account of site criteria for nuclear power plants

In 1950 the then Reactor Safeguards Committee under the old AEC prepared a report (AEC, 1950) proposing the creation of a so-called *exclusion radius* around a nuclear reactor where residences were not permitted. It was also proposed that such a radius, measured in miles, be 1/100 of the square root of the reactor power measured in thermal kilowatts, i.e. $R(\text{miles}) = \frac{1}{100}\sqrt{P(\text{kW})}$; the proposal was based on the assumption that the reactor will undergo a reactivity excursion, melting the core and rupturing the coolant system with the fission products escaping freely to the environment. The application to such a rule of thumb to nuclear power plants designed in the 1960s and 1970s led to an unacceptably large exclusion radius.

After realizing that the proposed rule of thumb was not appropriate to the many medium-sized demonstration reactors under consideration and after accepting the principle that reactor containment was better than isolation, in 1959 the AEC published in the Federal Register new proposed site criteria. The new site criteria maintained the concept of exclusion radius, now called exclusion distance, which depended not only on thermal power but also on the design features, mainly the inclusion of a containment system, and the site characteristics. For the large power reactors then considered the accepted exclusion radius varies from ½ to ¾ mile. It was also determined that beyond the exclusion radius population density should be small and there should be no large cities within 10 to 20 miles. In this preliminary document the main site parameters based on seismology, meteorology, geology and hydrology were also defined and considered.

After several intermediate steps, such rules were perfected and consolidated in a new document published in 1962 under the title *10 CFR Part 100 Reactor Site Criteria*, which has been maintained up to now. The new document consolidated the concept of exclusion area and created two additional concepts: a *low population zone* surrounding the exclusion area containing residents 'the total number and density of which are such that there is a reasonable probability that appropriate protection measures could be taken in their behalf in the event of a serious accident', and a *population centre distance* determining the minimum acceptable distance to 'the nearest boundary of a densely populated center containing more than about 25,000 residents'.

Methods to determine the corresponding radius and distances were also published by reference to a Technical Information Document (TID-14844) developed by DiNunno and co-workers (DiNunno *et al.*, 1962) based on a hypothesis regarding the consequences of the maximum credible accident, a concept that was introduced in 1959 by Dr Clifford Beck, a notorious regulator within the AEC, and on limiting radiation doses to the population (Beck, 1959). These ideas and concepts are maintained today and have had a deep influence in other countries; moreover they constitute one of the pillars for the safe design of most of the currently operating reactors. A full account of these historical developments has been published by Okrent (1981).

18.6.2 The present United States reactor site criteria

The NRC 10 CFR Part 100 titled *Reactor Site Criteria*, dated 1962, last amended in 1996, is divided into two subparts applicable to reactors built before 10 January 1997 and for site-related applications submitted on and after such date. The date separation is due to the creation of the new combined construction and operation licence (COL), while maintaining the previous system requiring separate construction and operation licences. In both cases a site permit is considered. Regarding population distribution the regulations require that every site must have an exclusion area and a low population zone and a population centre distance.

The *exclusion area* is formally defined as:

> 'that area surrounding the reactor, in which the reactor licensee has the authority to determine all activities including exclusion or removal of personnel and property from the area. This area may be traversed by a highway, railroad, or waterway, provided these are not so close to the facility as to interfere with

normal operations of the facility and provided appropriate and effective arrangements are made to control traffic on the highway, railroad, or waterway, in case of emergency, to protect the public health and safety.'

The *low population zone* is formally defined as:

> 'the area immediately surrounding the exclusion area which contains residents, the total number and density of which are such that there is a reasonable probability that appropriate protective measures could be taken in their behalf in the event of a serious accident.'

The *population centre distance* is defined as:

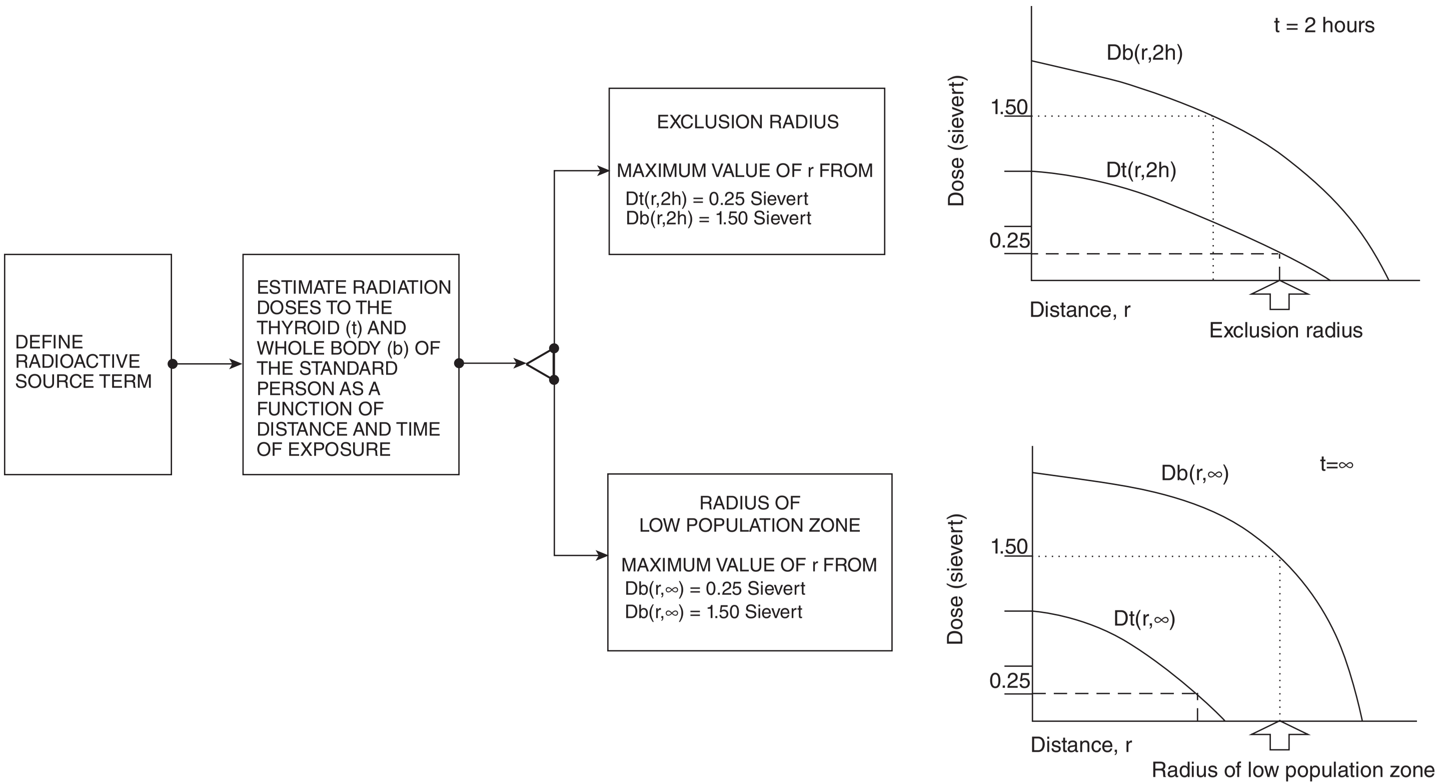
> 'the distance from the reactor to the nearest boundary of a densely populated center containing more than about 25,000 residents.'

It is added that the population centre distance:

> 'must be at least one and one-third times the distance from the reactor to the outer boundary of the low population zone.'

For currently operating reactors 10 CFR Part 100 makes reference to the TID-14844 document already mentioned (DiNunno *et al.*, 1962) to determine the values of the defined areas and distances. In the calculations the radioactive source term released to the atmosphere is based upon a major accident that would produce potential hazards 'not exceeded by those from any accident considered credible'. It is assumed that the whole reactor core will melt, releasing 100% of the noble gases' radionuclides and 25% of iodine radionuclides, in three different forms: 22.75% as elemental iodine; 1.25% as particulate iodine; and 1.0% as methyl iodide, a molecule observed in experiments with limited radiological importance. It is also supposed that the containment system will remain intact, releasing radioactive products at the containment's expected demonstrable leak rate; it is also assumed that containment spray and pool pressure suppression systems will function as designed and the release will be dispersed under assumed meteorological conditions.

Under such circumstances are calculated the whole body dose and the dose to the thyroid from the inhalation of iodine isotopes received by the standard man, standing in the axis of the radioactive plume, as a function of the distance from the release point for the first two hours after the onset of the release and during the total release time. The radius of the exclusion area is the maximum distance at which the assumed person will receive during the first two hours a whole body dose of 25 rem (equivalent to 0.25 sievert in SI units) or a thyroid dose of 150 rem (equivalent to 1.5 sievert in SI units). The radius of the low population zone follows the same criteria, but the reference dose will be received during the whole release time. The methodology is illustrated in Fig. 18.2. For a LWR of 1 GW of standard



18.2 Demographic site parameters.

design the radius of the exclusion area varies from 750 m to 1200 m, the low population zone from 3 to 5 km and the distance to population centres from 4 to 7 km.

The guide clearly indicates that the reference doses must not be considered as acceptable emergency doses to the public; they are only reference values for the purpose of defining the population distribution site parameters. The 0.25 sievert value for the whole body dose corresponds numerically with the once in a lifetime accidental or emergency dose for radiation workers which is generally disregarded in the person's radiation status.

The demographic site parameters should not be taken as reference to emergency planning, which normally covers up to 30 km from the plant and could be further extended and adjusted to any real situation. The basis of the calculation assumes the total meltdown of the reactor fuel, but the containment function responded as foreseen in the design considerably reduces the source term and its composition. In the 1979 TMI-2 accident 20% of the core melted, but the containment and corresponding safety safeguards responded as expected and only a preventive limited evacuation was considered necessary. In the 1986 Chernobyl-4 accident the nuclear core, the core pressure containment and the conventional building were destroyed by the vapour and hydrogen explosions, the fuel was dispersed and melted in the open air, releasing radioactivity over 10 days, and a large number of people far from the plant were evacuated or permanently relocated. In the 2011 Fukushima-1 accident three nuclear units were affected whose cores partially melted, but the containment and related safeguards only partially complied with their functions, resulting in substantial releases to the atmosphere of noble gases and volatile iodine, tellurium and cesium radionuclides. A large amount of contaminated water was also released to the sea. At the time of this writing, people were evacuated up to 20 km from the plant and recommended to take shelter at up to 30 km, for long periods of time.

18.7 References

AEC (1950), *Summary Report of the Reactor Safeguards Committee*, US Atomic Energy Commission, report WASH-3, Washington DC, AEC.

Beck C K (1959), Safety factors to be considered in reactor safety, in *Proc. VI Ressegna Internazionale Elettronica e Nucleare*, Rome.

DiNunno J J, Anderson F D, Baker R E, and Waterfield R L (1962), *Calculation of distance factors for power and test reactor sites*, US Atomic Energy Commission, TID-14844, Washington DC, AEC.

IAEA (2002a), *External human induced events in site evaluation for nuclear power plants*, Safety Guide, IAEA Safety Standards Series no. NS-G-3.1, Vienna, IAEA.

IAEA (2002b), *Dispersion of radioactive material in air and water and consideration of population distribution in site evaluation for nuclear power plants*, Safety Guide, IAEA Safety Standards Series no. NS-G-3.2, Vienna, IAEA.

IAEA (2002c), *Preparedness and response for a nuclear or radiological emergency*, Safety Requirements, Safety Standards Series no. NS-R-2, Vienna, IAEA.

IAEA (2003a), *Site evaluation for nuclear installations safety*, Safety Requirements, Safety Standards Series no. NS-R-3, Vienna, IAEA.

IAEA (2003b), *Meteorological events in site evaluation for nuclear power plants*, Safety Guide, IAEA Safety Standards Series no. NS-G-3.4, Vienna, IAEA.

IAEA (2003c), *Flood hazards for nuclear power plants on coastal and river sites*, Safety Guide, IAEA Safety Standards Series no. NS-G-3.5, Vienna, IAEA.

IAEA (2005), *Geotechnical aspects of site evaluation and foundations for nuclear power plants*, Safety Guide, IAEA Safety Standards Series no. NS-G-3.6, Vienna, IAEA.

IAEA (2007a), *Milestones in the development of a national infrastructure for nuclear power*, IAEA Nuclear Energy Series, no. NG-G-3.1, Vienna, IAEA.

IAEA (2007b), *Arrangements for preparedness for a nuclear or radiological emergency*, Safety Guide, IAEA Safety Standards Series no. NS-G-2.1, Vienna, IAEA.

IAEA (2009), *Volcanic hazards in site evaluation*, Document Preparation Profile (DPP) 405, Vienna, IAEA.

IAEA (2010), *Seismic hazards in site evaluation for nuclear installations*, Specific Safety Guide, IAEA Safety Standards Series no. SSG-9, Vienna, IAEA.

INSAG (1999), *Basic safety principles for nuclear power plants*, 75-INSAG-3 Rev. 1, INSAG-12, Vienna, IAEA.

Okrent D (1981), *Nuclear Reactor Safety. On the history of the regulatory process*. First printing, University of Wisconsin Press, Madison, WI, and London.

19
Bid invitation in nuclear power plant procurement

A. GONZÁLEZ, Empresarios Agrupados, A.I.E., Spain

Abstract: The Bid Invitation Specification (BIS) is a crucial element of the procurement process of a new nuclear power plant. The BIS documents constitute the basis used by the prospective bidders to prepare their bids for the nuclear power plant contract. In the BIS documents, the owner provides information regarding the project, instructions for the bidding process to be followed up to contract award, bid evaluation criteria, bid structure and contents, scope of supply requested, main project schedule milestones, technical requirements, as well as commercial, contractual and financing conditions applicable to the supply. It should be noted that the scope, structure and contents of the BIS documents largely depend on the contract approach selected by the owner for the project.

Key words: bid invitation specifications, BIS for nuclear power plants, bidding for nuclear power plants, owner specification for nuclear power plants, acquisition of nuclear power plants.

19.1 Introduction

The decision by an electric utility (hereinafter, the owner) to build a nuclear power plant, as part of the long-term nuclear power programme in a given country, usually follows a feasibility study carried out to provide the utility management and the relevant authorities of the country with the necessary information on which to base the decision to go ahead with the project and which demonstrates that the project is actually viable from the technical, economic and financial points of view.

Once the owner takes the decision to build a new nuclear power plant, two main phases of project implementation follow:

- The acquisition phase
- The construction phase.

The acquisition phase, sometimes referred to as the preconstruction phase, typically includes the following main activities:

- Feasibility study
- Site evaluation and selection
- Preparation of the Bid Invitation Specifications (BIS)

- Request for bids from prospective vendors
- Bid preparation by bidders
- Technical and economic bid evaluation
- Selection of the successful bidder (i.e., the vendor or supplier) and contract negotiation
- Contract signature.

The preparation of the Bid Invitation Specifications (BIS) is one of the most important preconstruction activities to be carried out by the owner during the plant acquisition phase. This chapter focuses on the preparation by the owner of the BIS as the key document by way of which he provides the bidders with information about the project, instructions on how to prepare and submit their bids, the scope of supply he wishes to purchase and the requirements of all types (i.e., technical, commercial and contractual) that he wishes to impose on the vendor for delivery of the plant.

19.2 Contracting approach and bid invitation specifications

19.2.1 Selection of the contractual model

At the beginning of the project implementation phase, one of the most important decisions that a new owner will have to make will be the selection of the contractual model under which the future nuclear power plant is going to be purchased.

Indeed, the contractual approach determines how the project management, design, equipment procurement, construction and commissioning management will be organised, and the extent to which the owner will be involved in these activities. It also establishes the distribution of risks and responsibilities between owner and vendor, for the successful outcome of the project.

The contractual model selected by the owner will have a significant influence on the structure and contents of some of the BIS documents, more particularly those dealing with scope of supply, project implementation and draft contract.

In the past, one of the following contractual approaches has usually been adopted for nuclear power plant acquisition:

1. *Turnkey contract*. A single supplier or a consortium of suppliers takes full responsibility for the delivery of the complete plant, ready for operation. The turnkey contractor therefore has complete responsibility for carrying out all phases of the project, from project management, engineering and design to procurement, construction, testing and commissioning.

2. *Split package contract*. Overall responsibility for the supply of the plant is divided among a reduced number of contractors. The owner places separate contracts for different portions of the plant (e.g. three or four large supply packages). Each of these contractors is responsible for the project management, procurement, construction, testing and commissioning of his own package or portion of the plant. The owner, on his own or with the assistance of an architect-engineering firm, takes responsibility for overall project management and integration of the design, construction and commissioning of the various packages. Following are some typical split-package contracting approaches, according to the number of packages the plant is divided into:
   - Two-package approach: the nuclear island (NI) is contracted separately from the turbine island (TI).
   - Three-package approach: package 1 corresponds to NI without civil works; package 2 is the TI without civil works; package 3 corresponds to the civil works for the NI and TI, contracted directly by the owner.
   - Three-package approach: package 1 is the NI and TI without civil works; package 2 is the BOP outside NI and TI without civil works; package 3 consists of the civil works.
3. *Multiple package approach*. The owner, on his own or with the assistance of an architect-engineering firm, assumes full responsibility for the engineering and design of the plant, as well as for the overall project management, equipment procurement, and plant testing and commissioning. The owner issues a call for tenders and places an order for the nuclear steam supply system (NSSS) and turbine-generator packages, based on which he develops the engineering and design of the complete plant, usually with an architect-engineering firm. He then issues a large number of contracts, with specifications prepared by the architect-engineer, to mechanical and electrical equipment vendors (e.g. for piping, valves, pumps, heat exchangers, electric motors, switchgear, instruments, and controls) and to construction and erection contractors at site. Sometimes also referred to as 'contract by components', this contracting approach has been extensively used in several industrialised countries by owners with experience in the handling of nuclear projects and in the direct management of other types of large complex projects such as fossil-fired power plants.

The choice of the preferred contractual approach depends on a variety of factors, some of which are listed below:

- Owner experience and knowledge in project management of similar projects, such as large fossil-fired power plants
- Local conditions available in the user country, including engineering, construction and erection capabilities, national infrastructures, qualified

human resources, and whether a single nuclear unit or a series of them are planned for the user country
- Experience in the user country of a pool of reliable equipment manufacturers and contractors with experience in the different contractual approaches
- Project costs, competitiveness and risks associated with each contractual approach
- Financing requirements and risk exposure perceived by lenders, depending on the contractual approach under consideration.

Regardless of the contractual approach finally selected, the owner will have to work closely with his project management team. Although substantial, the owner's involvement required under a turnkey contract is smaller than that required under other contractual approaches, and basically concerns project execution follow-up and contract administration and control, until the plant is turned over to him. The owner's involvement, risk and responsibility are greater in the split-package approach, and are considered to be maximum in the case of the multiple-package scheme (contract 'by components'). The degree of direct owner involvement also depends on the scope of work assigned to the architect-engineer assisting the owner.

Non-turnkey contracts have been largely used in countries where there is sound experience in large industrial projects. However, in countries that do not have experience in the handling of large complex projects or in heavy construction work, the turnkey approach seems the most suitable contractual model for the supply of a complete plant, and all the more so for owners from user countries planning to build their first nuclear unit. A good turnkey contract minimises the owner's risks regarding cost overruns, construction schedule, quality of the work and plant performance. Moreover, a turnkey contract for the first nuclear unit(s) would constitute a good learning exercise towards gaining experience, and provide a basis for selecting other contract approaches involving greater owner and local participation, in the event of building more units in the country.

19.2.2 Owner–supplier collaborative approach for turnkey contract

One contracting approach that is currently gaining acceptance in technology holder countries and in countries with nuclear experience planning to build new nuclear units is direct negotiation of the contract between the owner and a preselected bidder or a reduced group of prospective vendors. These direct negotiations may first take place with two (or three at the most) short-listed bidders to keep competition going at the onset of the vendor selection process and provide sufficient time to evaluate their bids and select one successful bidder as the prospective vendor with whom negotiations will be pursued up to contract signature.

The main objective of this direct, collaborative, 'open book' negotiating approach is to simplify the bid evaluation and vendor selection process, to minimise contingencies taken by the bidder, and to achieve a reasonable share of the economic risks between owner and vendor.

Under this direct negotiation approach, the owner selects one or more technologies and the corresponding prospective vendors, either according to his own preferences as an electric utility or following a technology assessment process. The owner prepares a complete set of BIS documents and invites bids from one or two preselected bidders. If bids are requested and received from more than one bidder, the owner undertakes a preliminary technical and economic bid evaluation, and then starts direct negotiations with the bidders, to agree on the technical aspects, scope of supply, terms and conditions, price, and other commercial conditions.

Following a period of time sufficient to establish which of the short-listed bidders has submitted the bid that is most advantageous to the owner, a prospective successful bidder is retained and full negotiations are undertaken with him until contract signature. Under this approach, the reduced number of preferred bidders are aware that they have to submit a competitive bid and that the competition remains valid during the first stage of the negotiations (during the owner's evaluation of the bids).

To reduce contingencies for the bidder and set the framework of the collaborative, 'open book' negotiation approach, the BIS requests bidders to identify in their bid which scope of supply packages or items are quoted as firm price subject to escalation under a lump sum offer, and which packages or items are quoted as non-firm prices (e.g. unit prices, costs with multipliers to be applied to the costs, time and material prices, etc.). These are prices to be converted into firm prices through a negotiation process initiated after a first bid evaluation. The owner then launches a direct, collaborative negotiation process, first with the reduced number of preselected bidders, and later on with the bidder finally selected, eventually to conclude a price agreement on the highest possible number of packages and/or items initially quoted in the bid as non-firm prices to convert them to firm prices subject to escalation, or to target prices subject to a 'gain and pain' scheme of incentives, to incorporate them into the lump sum portion of the contract.

If the price for a certain scope item could not be converted into a firm price and, therefore, is still open by the time of issuing the Final Notice To Proceed (FNTP), the negotiating parties will try to agree on rules of application of scope items quoted, for example, as unit prices to actual and reliable bills of quantities when they become available during the detail design completion process. Agreements may also be concluded on rules for items that were quoted at an estimated cost with an associated multiplier (or multipliers) at the time of bid submittal, which will be applicable when the time comes to purchase that specific scope item according to the project procurement schedule. These multipliers shall also be negotiated prior to contract signature, as they could typically cover contingencies, risk reserves, and margins to be applied to the actual cost of the scope item when this is determined.

Once a total price structure (lump sum fixed/firm price and non-fixed/firm price portion) has been agreed by both parties, as well as any other open points regarding technical requirements, scope of supply, schedule, and terms and conditions, the vendor is requested by the owner to convert the original bid completed with all the agreements reached during the collaborative 'open book' negotiation process into a final engineering, procurement and construction (EPC) proposal with the agreed price, scope, schedule and contract terms and conditions. The corresponding EPC turnkey contract is then established with as many of the scope packages as possible quoted as a fixed or firm price subject to escalation.

The direct collaborative 'open book' owner–vendor negotiation approach may be greatly facilitated by the existence of a standard plant design that is prelicensed in the country of origin of the technology, and counting on a high percentage of completed detail design. This approach also enables limiting the bidder's contingencies and the sharing of financial risks between owner and vendor, which corresponds to today's demand from the industry for the new and future nuclear power plant projects.

Finally, it should be noted that the owner requires a solid bid evaluation and negotiating team to implement this type of approach. This team may feature experts from the owner's organisation, with external support from an architect-engineering company with experience in the engineering, design, procurement and construction of NPPs.

19.3 Basis for preparation of the bid invitation specifications

19.3.1 Main decisions to be taken by the owner

The owner must take a number of decisions and make available important information to the team in charge of preparing the BIS before the latter can begin its work. Some of the most relevant information is the following:

- Owner (purchaser) identification
- Contractual approach
- Reactor type
- Number of units
- Power range (MWe) per unit and/or for the whole plant

- Plant location
- Site data
- Applicable codes, standards and regulatory requirements
- Cooling water system type
- Power grid characteristics
- Project schedule, including key milestones
- Licensing requirements and process to be followed
- Financing requirements of the project
- Scope of supply reserved to the owner
- Nuclear fuel (if required to be supplied by the plant vendor), and number of reloads to be supplied
- Quality management system requirements
- National participation policy
- Technology transfer objectives.

19.3.2 Planning and scheduling

The preparation of the BIS is a complex undertaking that requires the integrated contribution of a multidisciplinary group of experts covering the various disciplines involved in a nuclear project (e.g. licensing, nuclear safety, nuclear, mechanical, electrical, instrumentation and control (I&C), civil–structural, procurement, construction, commissioning, operation and maintenance, quality assurance (QA), legal, contracting, commercial, financing). Depending on the experience available, the owner's organisation and the availability of the required resources, the BIS may be prepared either by the owner's own personnel or with the assistance of an experienced outside architect-engineering (A/E) or consultancy company.

In the event that an external A/E or consultancy company is used, it should act in an advisory role. This means that it is highly recommendable that a parallel owner's team supervise, review and follow up the work performed by the external companies assisting the owner with the BIS preparation and take final responsibility for the decisions taken.

The composition of the team preparing the BIS will depend on the contractual approach. For a plant to be contracted on a turnkey basis, a team composed of 20 to 30 experts should be sufficient. The more segregated the procurement approach for plant acquisition, the more effort will be required on the part of the team. If the plant is contracted using large, split-package contracts (e.g. the NI separate from the TI), however, although the process would take longer, the resources required would not be substantially greater.

Six to eight months is a reasonable period for preparing the BIS. This would include the preparation of BIS criteria by the owner, preparation of several drafts of the BIS documents by the external A/E, review of the draft documents by the owner, and incorporation of the owner's comments into the documents by the A/E. This process is completed by a final review and approval of the complete BIS by the owner's management in time for issuing the BIS to prospective vendors.

19.3.3 Practical recommendations

Some practical recommendations for facilitating the preparation of the BIS are indicated below:

- Develop a solid multidisciplinary team, organised under the direction of a Project Manager, composed of experts who preferably have experience in nuclear projects. If this is not possible in certain areas, they should at least have experience in conventional power plant or large industrial projects.
- Prepare a detailed schedule showing all activities to be carried out and documents to be prepared, issued and reviewed by the different participants.
- Preferably, the BIS preparation team members should work in the same building or close to one another to ensure better integration of the work.
- Prepare the BIS documents using modern information technology (IT) tools in order to facilitate document production, revision and control. It is important to track and control the comments from the different participants and the changes made in the different versions of the BIS documents.
- The owner should always be in control of the work performed by the external A/E and/or specialised consultant (if any).
- The national nuclear regulatory authorities should be made aware of the work being done to prepare the BIS and be invited to provide their comments to the licensing, nuclear safety and other technical requirements specified in the BIS.
- It is advisable to hold periodic meetings with prospective bidders to review with them the BIS preparation approach being taken and to get their feedback.
- Existing reactor vendor standard plant designs, when available from the prospective bidders, should be taken into account during preparation of the BIS to ensure that the technical requirements set out in the BIS are realistic and based as much as possible on designs that are actually available on the market.
- It is of the utmost importance to have performed a comprehensive and detailed site study beforehand, so that reliable and complete site data (seismic, geological and environmental conditions, hydrology, cooling water characteristics, grid information, population distribution, social and industrial development) is available. This will enable the bidders to

prepare better bids and subsequently, to proceed more effectively with the plant design phase.
- To facilitate future bid evaluation work by the owner, special attention should be paid in the BIS to giving instructions to the bidders regarding the required structure and contents of their bids. This will ensure that all of the bids have similar structures, contents and information, which makes it easier to review and locate information in the bids.

19.4 Purpose, structure and contents

The BIS is the owner's specification for the plant he intends to purchase. The main purpose of the BIS is to provide the bidders (that is, the prospective vendors) with the necessary information to prepare their bids. It is through the BIS that the owner informs the bidders regarding the following:

- The scope of supply he expects to be offered
- The technical requirements in terms of plant design, procurement, construction, commissioning, operation and maintenance
- The manner in which he wishes the project to be implemented throughout the various execution phases
- The commercial and contractual terms and conditions he wishes to agree on with the successful bidder
- The structure, organisation and extent of technical, commercial and other information he expects to receive with the bid, to facilitate his evaluation and understanding of what is proposed by the bidder.

When organising, structuring and drafting the BIS documents, an important aspect to keep in mind is that they will serve as the basis from which will be developed the documents that will later constitute the contract between the owner and the successful bidder. Therefore, when preparing the BIS, one should always look ahead to how the contract documents will be organised and structured.

The BIS structure and contents very much depend on the contractual approach selected and the scope of supply requested by the owner. However, no matter which contractual model and scope of supply are chosen by the owner, and notwithstanding the bidding process the owner intends to follow (e.g. competitive bidding or direct negotiations with a single bidder), it is essential to prepare BIS documents that are specifically targeted at the particular circumstances of the project, describing the owner's requirements and conditions for plant delivery, providing the bidder with the information he requires to prepare his bid, and outlining to the bidder the information expected from him in the bid for a fuller understanding of what is being offered and for easier bid evaluation. There are many ways to structure the information to be included in the BIS, and any reasonable one is acceptable, as long as the information is complete.

Following is an example of how the BIS may be structured for a plant that the owner has decided to purchase complete (i.e. including nuclear island, turbine island and balance of plant) under a turnkey contract approach (i.e. including engineering and design, equipment supply, construction and commissioning). The BIS contents are organised into a number of separate documents, each dedicated to a specific topic, as can be seen below:

	BIS Documents
LI	Letter of invitation
IB	Instructions to bidders
SS	Scope of supply
TR	Technical requirements
NF	Nuclear fuel
PI	Project implementation
DS	Technical data sheets
DC	Draft contract
CC	Commercial conditions
FR	Financing requirements

The following sections present an overview of each of the above-indicated BIS documents. This tentative BIS table of contents, built around a complete plant under turnkey contract, may also be used as a guide in drafting up the BIS documents when the plant is to be purchased by large split packages (e.g. nuclear island separate from the turbine island), each of which could be contracted on a turnkey basis, or even for a multi-package or 'by components' approach under direct management of the owner. The BIS structure could be basically the same, but the contents of each document would have to be tailored according to the specific scope of supply and contractual approach selected by the owner.

19.5 Letter of invitation

The cornerstone of the BIS and written by the owner, the letter of invitation (LI) is addressed to the potential bidders, inviting them to submit their bids and briefly stating:

- Who is the owner company inviting bids and declaring its intention to proceed with the project
- Project name, reactor type(s), net power output range, number of units acceptable to the owner

- Plant location and main site characteristics (e.g. condenser cooling water method required)
- Contracting approach and scope of supply (base scope and options)
- Whether or not financing is to be arranged by the bidder
- Project scheduling forecast, and bid preparation and submittal schedule
- List of BIS documents enclosed with the LI
- Key BIS aspects that may be of special import to the owner and subject to special care from the bidders
- Request that the bidders formally notify, in writing and before the indicated date, of their intention to bid.

The LI should be brief (2–3 pages maximum) and avoid indicating details that are already covered in other BIS documents.

19.6 Instructions to bidders

Instructions to bidders (IB) provides clear instructions regarding bid preparation and submittal, description of the bidding procedure to be followed, establishment of the bid structure and contents, identification of the owner's evaluation criteria for the bids and description of the process of award to the successful bidder, leading to the signature of the contract. It is recognised that the IB document largely depends on the contracting model adopted by the owner.

The information provided and subjects covered by the IB document could be organised as outlined below:

1. *Introduction*. An introductory paragraph on the purpose of the IB document, followed by a short description of the following:
   - Purpose of the bid, name and location of the plant, reactor type(s), number of units, range of power output (MWe) for each unit that are acceptable to the owner
   - Summary information on the owner of the plant, who will be entering the contract with the successful bidder
   - Type of contract and implementation phases; project schedule and key milestones
   - Definition of the base proposal scope and of potential options/alternatives (as requested by the owner)
   - Financing arrangements, if financing is required by the owner
   - Requirements for bids submitted by a consortium or a joint venture (e.g. joint and several liability required among the partners)
   - Owner's requirements on the standard for business conduct throughout the bidding process and the implementation of the contract.

2. *Qualifications of the bidder*. This section should indicate the conditions required to qualify as an eligible bidder and submit a bid, the general requirements regarding bidder qualifications and the criteria applied to the selection of bidders.
3. *Bid invitation specifications*. An explanation of the purpose of the BIS and a short description of the different BIS documents, followed by indication of the procedure for the bidder to request clarifications regarding the BIS documents and for the owner to provide clarification or modification to the BIS documents.
4. *Bidding conditions*. Precise instructions in short separate paragraphs should be provided for each of the following topics: written notification of the intent to bid by bidders and deadline for this notification; inspection of site and obligation of the bidder to become familiar with the project and local conditions; confidentiality requirements during the bidding process; language to be used for bids and correspondence; communications between bidder and owner; system of units to be used; rejection of bids or cancellation of the bidding process by the owner; conditions for the modification and/or withdrawal of bids; no payment to bidders by the owner; procedure concerning deviations and exceptions to the BIS documents; owner requirements regarding bid commitment guarantee, parent company guarantee and contract performance guarantee.
5. *Bid structure and contents*. Specifies the structure, organisation and contents expected for each part of the bid to be submitted. A practical approach consists in structuring the bid into separate parts: Part I: Commercial Bid; Part II: Technical Bid; and Part III: Bidders Information and Qualification Documents.
6. *Delivery of bids, sealing and marking*. Presents the requirements for bid marking, sealing, packing and submittal, submission deadline, number of copies and information support, and procedure for opening the bids.
7. *Bid prices and validity*. Includes information related to price quotation, price breakdown, price escalation, payment schedule, taxes, bid validity period, and any other related aspect.
8. *Bid evaluation*. Presents the bid evaluation criteria to be applied by the owner, and procedures to be followed for the questions and answers process during the bid evaluation.
9. *Award of contract*. Informs the bidders on aspects such as shortlisting of preferred bidders, the notification of award, the execution and signature of contract documents, and notification to unsuccessful bidders.
10. *Forms and attachments*. Advice on aspects such as bid form, guarantee forms, price and price breakdown schedules, and list of subcontractors proposed by the bidder.

19.7 Scope of supply

19.7.1 Purpose

As its name indicates, the purpose of the scope of supply (SS) document is to define the scope of supply and services of the various participants in the delivery of the nuclear plant (i.e. vendors, the owner and other participants, as the case may be). The document should clearly describe the scope of each, the division of responsibilities (DOR), the respective limits of supply (terminal points) and the interfaces among project participants.

19.7.2 Contents

The power plant purchasing contract model selected by the owner largely influences the contents of the SS document; however, the structure of the document (i.e. its table of contents) basically remains the same, regardless of the contracting approach, and would organise the information as follows:

- Introduction
- Owner's scope of supply
- Bidder's scope of supply
- Other participants' scope of supply
- Definition of interfaces among project participants
- Division of responsibilities (DOR) tables
- Options.

Regardless of the contract approach selected (i.e. turnkey, split-package, multi-package), the SS document shall clearly specify the scope of supply and services that the owner assigns to himself, to the bidder and to the rest of project participants. To this end, the International Atomic Energy Agency (IAEA) Account System (IAEA, 2000) constitutes a good guideline. It consists of a list of all major items that make up the entire power plant scope of supply. It can be used as a reference to ensure that each scope item is assigned to one of the project participants. It can also assist in verifying the completeness of the scope of supply and ensuring that none of the scope items remains unassigned. There are other systems of account that can be used for the same purpose.

19.7.3 Variations according to contract approach

As regards the turnkey approach, no matter how detailed the description of the bidder's scope of supply in the SS document, it is highly advisable for the owner to protect himself with a 'completeness clause' clearly stating that the bidder is requested and shall therefore be committed to delivering a licensable and functionally complete plant, including all the services, structures, systems and components required for the plant to operate safely in accordance with the applicable codes, standards and regulatory requirements of the country and in compliance with the owner's technical requirements as laid out in the BIS.

When the owner has opted for the split-package or multi-package approach, redacting the SS document becomes a more complex undertaking to ensure that each plant scope item is clearly assigned either to the owner or to one of the package suppliers. Following are some practical recommendations:

1. A SS document should be prepared specifically for each individual large package (e.g. NI, TI, BOP, civil works) making up the complete plant. This SS document shall describe the scope of supply of the owner, that of the supplier and that of other participants for each specific large package.
2. As there will be several package suppliers, the overall responsibility of defining the scope limits (terminal points) for each package, of integrating all packages, of coordinating the various suppliers, and of managing and resolving interfaces among project participants remains with the owner.
3. In addition to the establishing the scope of supply and services of the owner, the SS document for each package shall clearly specify who is responsible for the performance of the following tasks referring to the overall project, which are not included in the scope of any of the individual packages:
   - Overall project management
   - Overall project schedule management
   - Overall site management
   - Overall plant commissioning management
   - Licensing support coordination of the entire plant
   - Management of interfaces between package suppliers
   - Overall plant performance guarantee.

   It is understood that each package supplier will be responsible for the project management, scheduling, construction and commissioning of his own package. Different package suppliers, as well as all other participants in the project, should be given a clear understanding of who will take overall responsibility for the management and integration of the various packages that make up the complete plant. The owner may decide to keep for himself the performance of these tasks for the entire project or he may hire an architect-engineering firm to perform these services. The latter, acting as the owner's engineer, will be responsible for overall management and integration of all packages on behalf of the owner.

4. Here again, the IAEA account system (IAEA, 2000) (or any other equivalent account system) provides guidance for the systematic checking of proper assignment to the owner, supplier or other project participant of all items that should be included in the scope of each package, and to ensure that no item has been overlooked.
5. It is good practice for the SS document to include a requirement of 'functional completeness' for the structures, systems and components constituting the package, that is, all piping and cables installed, all connections completed, and all fluids (oil, water, air, gases) delivered to the terminal points at the interfacing conditions agreed, which means that all systems and components should be fully operational.

19.7.4 Division of responsibility tables

A practical way of specifying the division of responsibilities (DOR) among project participants is to present it in table form, with a row for each scope item, listed as per the IAEA Account System (IAEA, 2000), for example. Depending on the contract approach selected by the owner, this can be done for the complete plant, separately for each of the main packages in a split-package contract, or for each of the main packages or items in a multi-package (i.e. 'by components' contract approach).

Table columns with specific headings allow allocation of responsibilities to the different project participants, who are designated by initials or acronyms indicated in the table layout (e.g. owner (o), plant/package supplier (s), civil works supplier (cv)). Following is a list of typical column headings to allocate the scope of supply and responsibilities:

- Input data
- Conceptual design
- Basic design
- Detail design
- Equipment procurement and supply
- Construction (civil works and erection)
- Testing and commissioning.

A last column entitled 'Remarks' provides space to include notes and clarifications to the responsibility allocation, when required.

19.7.5 Spare parts, special tools and consumables

The scope of supply for spare parts, special tools and consumable materials shall be specified in the SS document. The supplier shall be requested to provide all spare parts, special tools and consumables required for the installation, testing, commissioning and operation of the systems and equipment included in his scope of supply until final plant takeover by the owner. The bidder shall provide with his bid a list of recommended spare parts and special tools for the period extending up to takeover. Should additional spare parts or special tools that were not included in this list be required, for any reason, before plant takeover, they shall be furnished by the supplier at no extra cost.

The supplier shall also be requested to include in his bid another list of spare parts and special tools (indicating unit prices, quantity and delivery times) that will reasonably be necessary to ensure a number of years of normal plant operation (e.g. 3, 5 or 10 years). The owner may decide to request the submittal of separate lists, one for each specific period, to facilitate the evaluation and decision-making process.

The bidder shall also guarantee the availability and delivery of the spare parts for a reasonable number of years; in the event of a particular spare part or special tool becoming unavailable before the end of this period (e.g. production has been discontinued or the manufacturer has gone out of business), the owner should have the right to use all drawings and specifications relating to this item to procure it on the market.

19.7.6 Nuclear fuel scope of supply

A section of the SS document should be dedicated to specifying the scope of supply for nuclear fuel and associated services. Alternatively, this portion of the scope could be included in the nuclear fuel (NF) document of the BIS, which would be a comprehensive, self-supporting document dedicated entirely to the scope of supply and services, technical requirements and commercial conditions for nuclear fuel.

Standard practice is to request the following from the complete plant supplier (turnkey approach), from the nuclear island supplier (split-package approach) or from the NSSS supplier (multi-package approach):

- The first (initial) core loading, which should include:
  - Design, procurement and shipment of materials required for the fabrication and delivery to site of complete fuel assemblies and core components necessary for the first core loading of the reactor
  - Fuel services for the first core loading, including licensing documentation; transportation to site; supervision of fuel handling and loading into the reactor by the owner's staff; provision of procedures, special equipment and training for the owner's inspection and acceptance of fuel assemblies and core components at site; technology transfer to the owner for core design and reload safety evaluation; provision to the owner of all fuel assembly and core design documentation, including design basis, nuclear physics, mechanical and thermohy-

draulic design documents; provision to the owner of all necessary fuel data for him to achieve fuel procurement from third parties, if he should so decide in the future; and supply of all quality assurance and quality control manuals, procedures and records related to the nuclear fuel supply.

- Investment for reload batches. As an option, the bidder is usually requested to submit a proposal for the provision of a limited number of reload batches (usually two or three, sometimes more), sufficient for the owner's fuel specialists to familiarise themselves with the nuclear fuel and reactor core design and gain sufficient knowledge to decide whether to continue with the original fuel supplier or purchase it on the market from third parties. An alternative to requesting a specific number of reload batches consists in requesting the supply of reload batches necessary for a given period of operation (e.g. 4 or 5 years), after which familiarisation is expected to be achieved.
- Together with each subsequent reload, the supplier is normally requested to provide the associated fuel management services (e.g. core design, safety analysis, reload licensing).

19.7.7 Scope of supply for technology transfer

If the owner requires the supplier to provide technology transfer services in specific areas of the nuclear plant project, this should be indicated in a specific section of the SS document, which could include typical scope items such as training in accident analysis, training in probabilistic safety analysis, training in specific computer code applications and software modifications, to name a few.

19.7.8 Options

The SS document should have a section specifying the scope options that the owner requires from the bidder. The owner reserves the right to exercise each option, once it has been technically and financially evaluated. Options are a way for the owner to explore the convenience of including in the supplier's scope certain technical solutions for the structures, systems and components, or to modify the scope boundaries and make the decision at a later stage, once bidder information is available and has been evaluated.

This section should also make provisions for the inclusion of options proposed at the bidder's initiative. It may count on technical alternatives prepared by the bidders to the technical requirements specified in the BIS.

The owner should make it clear to the bidders that they should be committed first and foremost to complying with the BIS requirements, before any options be considered (be they requested by the owner or proposed by the bidder as technical alternatives). Otherwise these options or alternatives would be considered as exceptions to the BIS.

Bidders shall also be requested to submit complete technical information in their bids regarding options and alternatives, to facilitate evaluation by the owner.

19.8 Technical requirements

The technical requirements (TR) document is where the owner specifies the technical requirements applicable by the supplier to the plant design, licensing, procurement, construction, commissioning, operation and maintenance, and which shall be taken into account by the bidders in the preparation of their bids.

19.8.1 Main topics

The main topics addressed in the TR document are as follows:

- Applicable codes, standards and regulatory requirements
- Licensing requirements and procedures
- Site data, including geography and topography, geology, geotechnical and seismic information, methodology and environment (including site ambient and cooling water temperatures and humidity), demography, site access
- Design criteria and requirements for all project disciplines (e.g. civil–structural, mechanical, electrical, instrumentation and control, radiation protection, nuclear safety)
- Material requirements
- Requirements and specifications for plant structures, systems and equipment
- Power grid requirements
- Construction and erection
- Plant operation and maintenance
- Nuclear fuel requirements, including length of fuel cycle, spent fuel storage, refuelling operations
- Plant simulator requirements
- Personnel training.

19.8.2 Document structure

The TR document structure depends on the contract approach selected by the owner:

- Under a complete plant approach on a turnkey basis (NI + TI + BOP), it is sufficient to prepare one TR document, where the requirements can be organised as follows:

  - General requirements (applicable to the NI, TI and BOP)
  - NI technical requirements
  - TI technical requirements
  - BOP technical requirements.
- Under a split- or multi-package approach (e.g. NI, TI and BOP separate), preparing a specific TR document for each large package to be contracted separately (e.g. one for the NI, one for the TI and one for the BOP) results in a more practical and clearer procedure for the bidder. It is also possible to combine packages, for example in the case of BOP being contracted as part of the TI package, in which case a single document is prepared for both. Each of the separate TR documents should be self-standing, containing all the technical requirements applicable to the package, without need to refer to another TR document.

19.8.3 Preparation guidelines

Today the nuclear industry avails of two valuable documents that can be used as a reference when writing up the technical requirements for a new nuclear plant: the European Utility Requirements (EUR) for LWR Nuclear Power Plants in Europe (EUR, 2004, accessed 2011), and the EPRI Utility Requirements Document (URD) for Next Generation Nuclear Plants in the United States (EPRI, 2011). These are briefly outlined below.

European Utility Requirements (EUR)

The EUR document was developed by a number of European utilities, to establish a set of common voluntary requirements for the design of future LWR power plants in Europe. This document can also be applied to a wider, international market.

Some of the expected EUR application benefits are:

- Improved acceptance from the public and the authorities, achieved by using common technical solutions and common safety approaches
- Boosting nuclear energy competitiveness by controlling investment costs through the prescription of design standardisation and simplification, and by setting ambitious plant performance and maintenance cost reduction targets.

The EUR is structured into four volumes:

- Volume 1, 'Main policies and objectives', outlines the major objectives of the EUR organisation and the main policies laid down in the EUR document for future nuclear power plants, in aspects such as plant design, safety and licensing, standardisation, operational targets and economic objectives. It also summarises the most important requirements of Volumes 2 and 4.

- Volume 2, 'Generic nuclear island requirements', includes all the generic requirements and European utility preferences for the NI, which are not related to any specific design. Requirements are organised by chapters according to specific topics, as follows: safety requirements; performance requirements; grid requirements; design basis; codes and standards; material-related requirements; functional requirements: components; functional requirements: systems; containment systems; instrumentation and control and man–machine interface; layout rules; design process and documentation; constructability; operation, maintenance and procedures; quality assurance (QA); decommissioning; probabilistic safety analysis (PSA) methodology; performance assessment methodology; and cost assessment information requirements.
- Volume 3, 'Application of EUR to specific designs', is divided into a number of subsets. Each subset is dedicated to a specific design which is of interest to the EUR utilities. The volume contains a description of a standard NI, a summary of the analysis of compliance as compared to EUR Volumes 1 and 2, and, where needed, design-dependent requirements and preferences of the EUR utilities.
- Volume 4, 'Power generation plant requirements', outlines the generic requirements in relation to the power generation plant (i.e. the turbine island).

Utility Requirements Document (URD)

The URD presents a clear and comprehensive set of utility requirements for the next generation of nuclear power plants using LWRs in the USA. It was developed in the USA with the management and coordination of the Electric Power Research Institute (EPRI) and under the leadership of a group of American nuclear utilities. Some international utilities also took part in the development effort.

The URD consists of four volumes:

- Volume 0 is an 'Executive Summary'.
- Volume I summarises the US ALWR Program policy statement and top-tier requirements. Policy statements are formulated for the following key areas:
  - Simplification; design margins; human factors; safety; design basis versus design margins; regulatory stabilisation; standardisation; proven technology; maintainability; constructability; quality assurance; economics; sabotage protection; and good neighbour policy.

  Top-tier requirements cover the following areas:
  - General design requirements; safety and investment protection; plant performance; design process; constructability and economics.

- Volume II presents a complete set of both top-tier and detail requirements for evolutionary-type advanced light water reactors (ALWRs).
- Volume III provides a comprehensive set of top-tier and detail requirements for passive-type ALWRs.

The above Volumes II (Evolutionary ALWRs) and III (Passive ALWRs) each contain 13 chapters, as follows: (1) Overall requirements, defining common requirements applicable to a number of plant systems; (2) Power generation systems; (3) Reactor coolant system and reactor non-safety auxiliary systems; (4) Reactor systems; (5) Engineered safety systems; (6) Building design and arrangement; (7) Fuelling and refuelling; (8) Plant cooling water systems; (9) Site support systems; (10) Man-machine interface systems; (11) Electric power systems; (12) Radioactive waste processing systems; and (13) Turbine-generator systems.

19.8.4 Practical recommendations

One practical approach to preparing the TR document of the BIS for a light water reactor (LWR) plant consists in selecting either the EUR or URD as a reference and closely following the document selected as a model for organising and redacting the TR document. If the plant is to be built in Europe, the logical choice is to use the EUR as a reference; for a plant in the USA, it seems reasonable to follow the URD. The choice is no longer as clear-cut for other countries: the owner will have to decide which reference document (EUR or URD) is the most suitable to his own preferences and criteria as a utility, and to the regulatory requirements applicable in his country.

Neither the EUR nor the URD are actually specifications of technical requirements to purchase a nuclear power plant. They both constitute a set of functional requirements and design objectives with which the European and American utilities that are writing and promoting them would like new plants to comply.

The following procedure is suggested (note: if suggested by the author, it should be clearly specified; if taken from a reference, it should be included) to convert the EUR or URD into an actual set of technical requirements for BIS to purchase a new nuclear plant:

1. Choose which of the two (EUR or URD) will be used as a reference document.
2. Use the table of contents, structure and wording as a starting point.
3. When a given requirement, design criterion, objective, data or limit value in the text does not comply with the requirement of the country's regulatory authorities, replace or modify the original wording of the

reference document as necessary to reflect the applicable regulatory requirement.
4. Modify and/or complement the original wording of the reference document with the owner's requirements that reflect its own preferences and practices as a utility with experience in the construction, operation and maintenance of power generating plants.
5. Introduce additional technical requirements for structures, systems and equipment design, materials, fabrication and testing that are more in line with the actual specifications of the owner for a power plant, thereby transforming a reference document featuring basic design objectives and functional requirements into an actual procurement specification for the plant.

The above procedure is illustrated by the following example: the owner selects the EUR as a reference to prepare the TR document for a complete plant (NI + TI – BOP) to be supplied under a turnkey contract. The TR document would likely be organised as follows:

- Part 1: Main policies and objectives. This part closely follows EUR Volume 1 and specifies the main design, construction and operation policies and design aspects required by the owner at the overall plant level.
- Part 2: Nuclear island requirements. This part specifies the owner's technical requirements for the NI. It should follow the same structure and contents by chapters as EUR Volume 2. Some of the NI chapters (e.g. codes and standards, material-related requirements, components and systems, constructability) may be common to the NI (Part 2) and power generation plant (TI) in Part 4 and written as such, thereby avoiding the need to repeat the same requirements in Part 4.
- Part 3: Nuclear fuel requirements. This part contains all the technical requirements specifically addressing nuclear fuel supply and management services.
- Part 4: Turbine island requirements. This part includes the technical requirements specific to the power generation plant (that is, the turbine island). It may also include requirements for the BOP outside NI and TI. The structure and contents of Part 4 could follow those of EUR Volume 4. As mentioned above, some of the chapters in Part 2 may be common to the NI and TI; such commonality should be taken into account when writing the chapters for the NI.

19.9 Project implementation

In the project implementation (PI) document, the owner specifies the requirements for implementing the project. To this end, the PI document should describe the project management and organisational, quality and environmental management, project planning and scheduling, project risk evaluation, project control, engineering and design management, procurement and supply chain management, project documentation, information management system (IMS) and project communication requirements which the bidder is required to follow to carry out the contract.

Preparing a good and complete PI document will increase the probability of having a well-organised project. The following paragraphs outline the proposed contents for the PI document.

19.9.1 Project management and organisation

This describes the owner's project implementation model, the requirements for the project organisation and management manual to be prepared by the bidder, a description of the owner's organisation, the requirements for the vendor's organisation, the assignment of key organisational responsibilities, a description of the licensing process to be followed and a definition of licensing responsibilities. This section could conclude with a description of the risk management system to be applied to the project.

19.9.2 Quality and environmental management, occupational health and personnel safety

This section establishes the codes, standards and regulations applicable to the quality and environmental management system to be applied in the project. It should also require and provide instructions to the vendor for the preparation of his general quality assurance and environmental management plan (GQAEMP) for the project, structured and organised to cover the topics defined in the applicable quality and environmental codes and standards. The vendor should be required to identify, prepare and provide the project procedures applicable to each of these quality assurance and environmental (QAE) topics. The vendor shall be requested to enforce the application of the GQAEMP by his subcontractors. Other subjects to be covered in this section are the owner's requirements concerning quality audits, inspections and reports, and the disposition of non-conformances and of corrective actions in the project. Finally, this section should also contain the requirements for the preparation by the vendor of the occupational health and personnel safety plan for the project.

19.9.3 Planning and scheduling

This section should define the plant delivery schedule expected by the owner, indicating main milestones for the project, from contract signature and notice to proceed (NTP) through provisional acceptance of the plant and commercial operation date, up to final plant acceptance at the end of the guarantee period.

Other main subjects to be covered here are the planning and scheduling preparation criteria and schedule preparation software tools to be used; required scheduling levels (e.g. level 1, level 2 and level 3 schedules, each of them featuring an increasing level of detail and a progressive number of activities); integrated overall project schedule, as well as a schedule for each of the main project phases (e.g. design, licensing, equipment procurement, construction and commissioning phases).

19.9.4 Project control

The requirements for adequate project control by the owner should be established in this section. Subjects to be covered are schedule updating and periodic submittal procedures; project progress control and preparation guidelines for project progress reports, indicating the minimum information to be included and the frequency of submittal; control of project design criteria to be prepared by the vendor, and containing the project design basis, as well as functional and technical baselines, organised by project discipline (e.g. nuclear safety, mechanical, electrical, instrumentation and control, civil engineering); licensing and permitting process, covering both the nuclear licensing of the facility and the conventional permits to be obtained; project design changes and configuration control requirements; description of the vendor's system for procurement and material management, including vendor procedures for requesting bids for the purchase of equipment and materials, preparation of subcontract packages, award and administration of subcontracts, as well as indication of the procurement documentation required both from the vendor and his subcontractors (such as equipment specifications, inspection plans, test procedures, quality reports, manufacturing schedules); control and follow-up of subcontractors; and traceability system implemented to follow up on the procurement process, and how it should be supported by an information management system accessible to the owner.

19.9.5 Engineering and design management

Requirements regarding the supplier's engineering organisation, engineering and design process, design interfaces management, use of computer-assisted design (CAD) software and tools, project design manual, review and approval of design documentation by the owner and other aspects related to the management of the project engineering process will be specified in this section.

19.9.6 Procurement and supply chain management

This section should specify the requirements for the procurement process, formation of the supply chain for the project, use of subcontractors (i.e. from a list of owner-approved subcontractors), inspection plans, approval of manufacturing procedures, and project procurement procedures manual.

19.9.7 Project risk evaluation

This section should specify the requirements concerning the economic and financial aspects, as well as project scheduling and development, and other relevant information required by the vendor for the preparation and periodic updating of a project risk evaluation report, enabling the owner to assess the risk status of the project throughout its duration.

19.9.8 Project documentation

This section of the PI document includes requirements on the documentation to be submitted with the bid by the vendor; the list of project document types to be submitted to the owner's review and/or approval; the requirements for the preparation of the list of project documents, grouping documents by project phase (e.g. design, procurement, construction, testing and commissioning, plant operation and maintenance); a description of the owner's review and approval process for project documentation; document formatting and submittal requirements (e.g. electronic and/or hardcopy, format); specific requirements for the vendor's documentation concerning package plants; and final project documentation to be handed over to the owner by the vendor for project records and plant operation and maintenance.

19.9.9 Information management system

The information management system (IMS) describes the owner's minimum requirements to be used by the vendor during the design, procurement, construction, testing and commissioning stages of the project. This may include specific requirements, such as the use of certain software applications and databases to ensure compatibility with the owner's system and eventual transfer of project information for use during plant operation and maintenance, once construction is completed by the vendor. Special care should be taken in identifying which information in the vendor's databases is to be made accessible to the owner at all stages of the project, so as to facilitate project control and monitoring by the owner.

Another significant aspect of the vendor's IMS that requires close attention in this section of the PI document is the software tools that the vendor intends to use at the design stage for the performance of engineering and design activities such as 3D modelling of structures, preparation of HVAC ductwork and cable raceway layouts, production of piping and instrumentation diagrams (P&IDs), schematic and wiring diagrams, piping isometrics, engineering of component databases, and computer codes for engineering calculations, to ensure smooth transferral to the owner at the end of the project and future use in plant modifications and upgrades during the operation phase.

19.9.10 Project communications

This section is dedicated to laying out the owner's requirements for communications among project participants. The main topics to be addressed are requirements for written communications, correspondence filing and coding system, correspondence distribution criteria, record keeping, quality assurance requirements for the transmission of quality-related design data, and record of correspondence pending answer.

19.10 Technical data sheets

The technical data sheets (DS) document consists of a set of data sheets summarising the following information in a table:

- The main technical requirements specified in the TR, NF and PI documents
- The main technical data for plant structures, systems and components.

The table format usually presents the information in the following manner: in the left-hand column, the owner briefly sums up the key technical requirements for which a summary answer is requested from the bidder (to be included in the right-side column, left blank). This table is to be completed by the bidder directly and included in his bid.

It should be noted that data sheets are only a summary of technical features and data presented in an organised and systematic manner, for the purpose of obtaining, in addition to the bids: (a) answers from all the bidders organised in the same fashion; (b) a quick understanding of the compliance of each bidder with key requirements; and (c) a consistent tabulation of plant data to facilitate bid evaluation. Thus it is understood that more detailed information regarding plant design and features is to be found in the technical descriptions, drawings and data included by the bidder in other parts of the bid.

It is advisable for the technical data sheets to be organised into three sets, as follows:

1. Data sheets covering the main data at the overall plant level, to provide a quick overview of plant design parameters and features.
2. Data sheets addressing compliance with the main top-level design objectives as they are specified in the BIS and organised by project discipline (e.g. licensing and regulations; nuclear safety; civil–structural; NSSS; systems and equipment; turbine-generator and steam cycle; radwaste systems; electrical systems; I&C; BOP; project implementation).
3. Data sheets listing the main technical data (i.e. performance, design conditions, operating conditions, materials, quality classification, codes and standards, etc.) of main plant structures, systems and components. This third set of technical data sheets can be organised as follows: NSSS systems and components (e.g. reactor pressure vessel, steam generators, main coolant pumps); emergency core cooling systems (ECCS); reactor auxiliary systems, containment systems, nuclear fuel supply and handling; radwaste systems; plant auxiliary systems; electrical systems; I&C systems; turbine-generator and auxiliaries; steam-cycle systems (main steam, feedwater, condensate, etc.); and cooling water system.

19.11 Draft contract

19.11.1 General

The draft contract (DC) document constitutes the draft of the final contract proposed by the owner and which he intends to sign with the selected bidder. Again in this case, if the contract concerns the turnkey purchase of a plant (single-package approach), there will be one main contract; in the event of a multi-package (two or more) approach, a separate contract will be drafted for each package included in plant procurement.

The DC document should basically contain:

- The owner's proposed terms and conditions for the final contract, covering all the legal, administrative, organisational, technical, economic, financial and commercial aspects of the transaction that require agreement between the owner and the successful bidder in the final contract
- The identification of the 'contract documents', that is, all the documents that will form part of the final contract, listed in the order of precedence to be applied *vis-à-vis* one another in case of discrepancy.

Far from being a matter exclusively for engineers or for lawyers, the preparation of the terms and conditions should preferably involve a multi-disciplinary team of experts for the technical aspects, lawyers for the legal aspects, as well as experts for licensing and permitting, insurance and finance. Including the participation of a lawyer (or lawyers) familiar with the laws of the owner's country and with international contracting is especially advisable.

As regards nuclear fuel, a single contract may be devised for both the plant and the nuclear fuel, or separate contracts may be prepared for the plant supply and for the nuclear fuel. The latter approach is more frequent.

19.11.2 Contract documents

As seen above, the contract is made up of several 'contract documents' that should ideally be listed and defined in the DC document. A typical structure and contents of plant contract documents is given below:

1. Contract agreement.
2. Terms and conditions, as well as the following appendices:
   Appendix I: Price
   Appendix II: Price escalation formula
   Appendix III: Payment schedule
   Appendix IV: Contract guarantees
   Appendix V: Contractual project schedule
   Appendix VI: Advance payment guarantee
   Appendix VII: Contract performance guarantee
   Appendix VIII: Warranty period guarantee
   Appendix IX: Parent company guarantee
   Appendix X: List of approved subcontractors
   Appendix XI: Supplier's consortium/JV agreement (if applicable).
3. Owner's specifications, comprising:
   – Scope of supply (SS) document
   – Technical requirements (TR) document
   – Project implementation (PI) document
   – Technical data sheets.
4. Other complementary documentation agreed by the parties to form part of the contract.

In case of discrepancy between any of the contract documents, the following order of precedence shall prevail:

1. Contract agreement
2. Terms and conditions with their appendices
3. Owner's specifications
4. Supplier's technical bid
5. Supplier's information and qualification documents.

In principle, the preparation of terms and conditions for contracting a nuclear power plant bears a certain similarity to their preparation for purchasing a conventional fossil-fired or combined cycle power plant. The structure and contents are similar, although there are a number of issues that are specific to nuclear power and thus require special attention and treatment.

A typical table of contents for the terms and conditions to be prepared and presented to the bidder in the DC document of the BIS is shown below:

1. Introduction
2. Definitions and interpretation
3. General contract provisions
4. Mandatory law, requirements of the authorities, and codes and standards
5. Purpose of contract and scope of supply
6. Licensing
7. Quality and environmental management
8. Project documents
9. Contract price
10. Revision of contract price
11. Terms and schedule of payments
12. Payment execution
13. Contract variations
14. Confidentiality and intellectual property
15. Risk and title
16. Liabilities
17. Insurance
18. Project schedule and delays
19. Testing, commissioning and provisional takeover
20. Warranties and performance guarantees
21. Owner's acceptance and final takeover
22. Force majeure
23. Owner's personnel training
24. Rejection and termination of contract
25. Governing law
26. Settlement of disputes
27. Notices
28. Joint and several liabilities (when the supplier is a consortium or JV)
29. Contract assignment and subcontracting
30. Spare parts
31. Other miscellaneous conditions
32. Severability
33. Survival of obligations

34. Relationship of the parties
35. Entire agreement and contract amendments

    Appendices:
    - I. Price
    - II. Price escalation formula
    - III. Payment schedule
    - IV. Contract guarantees
    - V. Contractual project schedule
    - VI. Advance payment guarantee
    - VII. Contract performance guarantee
    - VIII. Warranty period guarantee
    - IX. Parent company guarantee (if applicable)
    - X. List of approved subcontractors
    - XI. Supplier's consortium/JV agreement (if applicable)

The bidder shall be requested in the BIS to specifically declare compliance with the proposed draft contract or to submit a list of exceptions and comments to it, to be discussed during contract negotiation and presumably leading to an agreement between the owner and the bidder regarding the final version of the contract.

19.12 Commercial conditions

The commercial conditions (CC) is the BIS document in which the owner establishes the information required from the bidder as regards prices, price breakdown, price escalation formulae, payment terms and schedule, and other commercial conditions for the scope of supply and services offered.

The owner must request all information regarding prices and commercial conditions to be provided in sufficient detail to facilitate the economic and financial evaluation of the bids and to serve as the basis for establishing the commercial conditions of the contract.

19.12.1 Prices and price breakdown

The bid prices quoted by the bidder for the scope of supply and services offered are usually referred to as 'base bid prices'. They can be fixed, firm, unit prices or just budgetary/estimated prices, which the bidder shall indicate in his bid.

A fixed price is binding on the bidder if it is accepted by the owner during the bid validity period. However, it is not subject to adjustment as a result of escalation and is based on the delivery of the item at the commercial operation date (COD) of the plant.

A firm price is also binding on the bidder if it is accepted by the owner during the bid validity period and is subject to adjustment as a result of escalation. The bidder shall include in his bid the escalation formula applicable to each firm price.

The bidders will be requested to submit a price breakdown in their bid. The level of price breakdown should be sufficient to enable financial evaluation of the bid and its comparison with other bids. The IAEA accounts system (IAEA, 2000) provides good guidance regarding price breakdown level. The bidder's price breakdown schedule should clearly indicate the kind of price associated with each scope package or item: whether it is fixed price not subject to escalation, firm price subject to escalation, unit prices, budgetary prices, or any other price category foreseen in the CC document. In summary, the bidder shall specify which part of the bid is quoted fixed price, which part is firm price with escalation and which parts are quoted in other price categories.

When it comes to price breakdown, it should be understood that bidders may be reluctant to provide a high level of detail in the segregation of the bid price. Scope packages or items quoted as fixed/firm prices should require no breakdown or a small one. A reasonable and practical level of price breakdown that can be requested from the bidder is to indicate the price for each major account (for example, for each two-digit or three-digit account) of the IAEA accounts system (IAEA, 2000).

For scope packages and items quoted as non-fixed/firm prices, the owner should request from the bidder a higher level of detail for the price breakdown, to set the basis for negotiations.

Following are some aspects to be considered when specifying the level of price information to be provided by the bidder:

- The price breakdown should make it easier to evaluate and compare bids.
- Price segregation should always permit the application of the different price escalation formulae for adjustment of the base price offered in the bid.
- Prices should distinguish the portion of the scope offered in foreign currency (or currencies) from that quoted in local currency, to enable evaluation of exposure to foreign currency exchange risk.
- Prices should indicate the portion (%) corresponding to local supply and services, to enable calculation of the local participation offered by the bidder.
- Prices for scope packages or items sourced from different countries could be quoted in the currency of the country of origin of the supply, in the currency in which the bidder wishes to be paid, in which case the portion (%) offered in each currency should be indicated to enable calculation of the owner's exposure to foreign currency exchange risk. The bidder may also be requested by the owner to quote everything in

a single currency, in which case the fluctuation in the exchange rates with respect to the common currency will be at the bidder's charge.

- Prices should preferably be presented in tabular form, with a row assigned to each scope item for which a segregated price is offered. Typical column headings of the price table are scope item number and description, price type (e.g. fixed/firm, unit price, budgetary price), percentage of price in foreign and/or local currency, and remarks (if necessary).

19.12.2 Price revisions

As indicated above, the base prices quoted may be fixed and not subject to escalation, or firm and subject to escalation, or any other type of non-fixed/firm price. The owner shall specify in the CC document of the BIS his requirements concerning the methodology and price adjustment formulae to be proposed by the bidder. Typically there should be more than one price adjustment formula. For example, there can be:

- One price adjustment formula for the revision of the base price of scope items associated with the delivery of services for which only labour cost indices will be used
- One (or more) price adjustment formula for the revision of base prices quoted for the scope of supply items involving manufacture or construction, for which both labour and material cost indices will have to be considered in the formula. More than one material cost index may also be included in the formula, when different categories of materials having differentiated cost variation over time are used in the manufacture.

The labour and material cost indices used in the price adjustment formulae shall be those published by an official institute of the country of reference and should have a long record of publication (at least 10–15 years). Should any index be discontinued, the index which officially replaces the discontinued one shall be applied from then on; when there is no index to officially replace the discontinued one, the owner and the successful bidder (the contractor) shall agree on the selection and application of another existing and widely recognised index that reflects as closely as possible the same items and provides results similar to those of the original index.

19.12.3 Terms of payment

In the CC document, the owner shall request the bidder to submit a payment schedule, clearly indicating the amount of the advance payment (if so requested by the bidder) and of the payments linked to the fulfilment of each milestone listed in the payment schedule. The owner may wish to specify in the BIS project milestones that should be complied with as a minimum to receive partial payment. Payment milestones should be linked to a representative delivery or measurable project progress, and should also ensure the owner that payments made are commensurate with actual project progress.

Payments to be made upon fulfilment of a milestone should be differentiated according to whether they are made in local or foreign currency.

Finally, the owner should establish in the DC document of the BIS all the detailed contractual conditions regarding payment, such as payment currency, advanced payment guarantee, criteria to be applied in case of delayed payment, failure to make a payment by the owner, disputes concerning payment, invoicing rules, form and place of payment after invoicing, and setting-off.

19.13 Financing requirements

If financing is required by the owner for the project, the owner may decide to arrange it directly with the financing institutions, or he may wish to request the bidders to submit proposals for financing arrangements together with their bids for plant supply.

The owner may have developed plans to finance the whole, a substantial part or just a portion of the project (for example, the foreign components of the bidder's scope of supply, including nuclear fuel). Whatever the plans and expectations of the owner regarding project financing, the BIS should clearly indicate whether financing is required to be provided or arranged for by the supplier.

The purpose of the FR document of the BIS is to specify:

- The scope and conditions of the financing required by the owner to be provided or arranged by the supplier
- The information that, as a minimum, shall be submitted by the bidder in his financing proposal to provide clear understanding of the financing conditions offered and facilitate the owner's evaluation.

The financing institutions (lenders) with whom the bidder has arranged the requested financing scheme shall present a complete financing proposal including all the information requested by the owner; this document shall be submitted together with the plant supply proposal prepared by the bidder.

The FR document should request the bidder to describe, in his financing proposal, the financing instruments and approach that he intends to use (such as export credit financing, co-financing, multi-country financing, and supplier's credit).

Export credit financing (buyer's credit) is a common approach in the financing of nuclear projects. The following paragraphs indicate the typical information that the owner should request in the financing proposal in case this financing approach is selected.

19.13.1 Typical financing information to be provided by the owner

The typical information that must be supplied by the owner in the FR document when the buyer's credit approach is applied is as follows:

- Name of project
- Country of project
- Buyer and borrower identification
- Guarantor (if any)
- Financing project description
- Scope of financing required. For example, if financing is requested only to cover the foreign contents of the scope of supply:
  - Portion of the bid scope and price for which financing is required
  - Financing of escalation and interest during construction
  - Financing of export credit agency (ECA) insurance premium
  - Financing of charges and fees

  Portions of the above-indicated items to be financed through a buyer's credit insurance of the ECA or the exporting country and portions to be financed through a commercial loan
- Currency(ies) of the credit
- Starting date of the credit (typically the day of provisional takeover)
- Repayment terms.

19.13.2 Typical financing information to be provided by the bidder

Typical information to be requested by the owner in the FR document for submittal with the financing proposal, for each financing source, when the buyer's credit approach is applied is as follows:

- Source of financing
- Amount of loan
- Currency(ies) of the loan(s)
- Arranger and agent
- Lender(s)
- Drawdown period
- Grace period
- Starting date for repayment

- Payment schedule
- Payments, amortisations and interest rates
- Insurance premiums
- Financing of charges and fees
- Expenses
- Taxes
- Governing law
- Other terms and conditions.

The requested validity period of the financing proposal shall be at least the same as that of the commercial proposal submitted by the bidder for his scope of supply.

19.14 References and further reading

EPRI (2011), *Utility Requirements Document (URD) for Next Generation Nuclear Plants,* Electric Power Research Institute, Inc., Palo Alto (California). Available from: http://urd.epri.com/ [Accessed 22 February 2011]

EUR (2004), *European Utility Requirements for LWR Nuclear Power Plants*. Volumes 1 to 4. Available from: http://www.europeanutilityrequirements.org/eur.htm [Accessed 22 February 2011]

IAEA (1993), *Financing arrangements for nuclear power projects in developing countries: a reference book*. Technical Report Series No. 353, IAEA, Vienna.

IAEA (2000), *Economic evaluation of bids for nuclear power plants*: 1999 edition, Technical Report Series No. 396, IAEA, Vienna.

IAEA (2007), *Managing the first nuclear power plant project*. IAEA TECOO-1555, Vienna.

IAEA (2009), *Common user considerations by developing countries for future nuclear energy systems*. Nuclear Energy Series No. NP-T-1.17, IAEA, Vienna.

IAEA (2010), *Invitation and evaluation of bids for NPPs*, Publication NG-T-3.10 (Draft version published), IAEA, Vienna.

20
Licensing for nuclear power plant siting, construction and operation

A. ALONSO, Universidad Politécnica de Madrid, Spain, S. K. SHARMA, formerly of Atomic Energy Regulatory Board, India and D. F. TORGERSON, Atomic Energy of Canada Ltd, Canada

Abstract: This chapter addresses the need for licensing of nuclear power plants, and how such licenses can be requested by an applicant and granted by a regulatory authority. The licensing process is country dependent, although based on the common principle that the applicant must demonstrate that the proposed nuclear power plant will comply with the established regulations, and that it will operate safely without undue risks to the health and safety of plant personnel, the population and the environment. During the construction and operational phases the regulatory authority ensures compliance with the the license conditions through evaluation, monitoring and inspection. The license may be a single document covering all the phases in the life of the plant, or a set of consecutive documents requested and issued for different phases, which may include design certification, site approval, design and construction, commissioning and operation, design changes during operation, life extension and, finally, decommissioning.

Key words: site license, construction license, commissioning license, operating license, decommissioning license, design certification, license renewal.

20.1 Introduction

Nuclear power plants and related fuel cycle installations and activities are built and put into operation because they offer advantages for the global need for electricity generation. However, these installations and activities have a potential to create radiation risks to the health and safety of the population, and to cause radioactive contamination of the environment. The need then arises to keep such risks under control and reduce them to acceptable levels, while maintaining the economic, environmental, and social advantages from such installations and activities. That goal can be reached by scientific understanding of the phenomena behind such risks and implementation of technical measures to overcome them. Although much knowledge has already been obtained and relevant technical progress has been made and put into practice, as in other similar cases, it has been considered necessary to establish a strict independent licensing system. This chapter discusses the meaning and purpose of licensing, and looks at the implementation of the licensing process for nuclear power plants.

Licensing of nuclear power plants is a well-regulated activity by which the potential licensee submits a proposal in accordance with specified requirements. A competent body of experts then verifies that safety provisions fully comply with the previously established safety requirements. A licensing authority makes its decision based on the safety assessment provided, as well as on other national requirements. There is a large variety of national organizational setups dependent on individual countries' legal infrastructures and practices; nevertheless, the licensing principles are equivalent. In some cases the expert body and the licensing authority are within a single organization, whereas in other cases the licensing authority, generally a government authority, is separated from the body of experts. Whatever the system, within this chapter, the body of experts and the licensing authority together are referred to as the Regulatory Body (RB).

There are some countries in which a single license, although divided into parts, covers all phases in the life of the plant, while others license each phase independently. In both cases, well-established steps or parts have been defined that include the siting, design and construction, commissioning, operation and dismantling of a plant. Some countries include design approval as a first step of licensing, as well as plant modifications during construction and operation. Some RBs also license several types of reactor operating personnel. This chapter includes examples of such approaches.

Both the applicants for a license and those who carry out the safety review need to have a good knowledge of the nuclear power plant (NPP) design and experience of relevant legal, scientific and technological issues. The applicants need to have a deep knowledge of the safety requirements and the technologies to demonstrate compliance with them. Safety reviewers have to be able to verify that compliance with the regulations has been adequately demonstrated. For the first units in new entrant countries, applicants may obtain help from reactor suppliers, while the safety reviewers should acquire the needed expertise from the RB of the country of origin of the NPP supplier, or from an experienced regulator that has licensed one or more NPPs employing the selected technology. In any case, adherence to well-proven designs is highly recommended. When new designs are employed, they should be thoroughly checked by analysis and testing. This chapter also describes the areas in which help from an experienced regulator can be gainfully used by new RBs of new entrant countries.

Safety should not only be achieved in design, siting and construction but also be maintained and improved during all modes of operation, including commissioning and decommissioning. To achieve this goal, there should be a strong safety culture and positive safety attitude on the part of the licensee, and an efficient and effective nuclear safety overview process by the RB with the capability of enforcement in case of deviations from the established requirements. Periodic self-evaluations, as well as peer reviews by national and international experts, like those conducted under the systems in practice by the International Atomic Energy Authority (IAEA) and the World Association of Nuclear Operators (WANO), are also helpful in achieving and maintaining a high level of safety. The responsibilities and major functions and activities to be performed by the licensees and the RB are addressed in this chapter.

20.2 The need for licensing

Operation of an NPP generates large quantities of radioactive material from the fission of nuclear fuel, and by neutron activation of reactor system fluids (during their passage through the reactor core) and of the structural materials in and around the reactor core. The radioactivity so generated must be confined or disposed of such that it does not cause undue radiation hazard to plant personnel, the public or to the environment. This objective is achieved by ensuring that the design, construction and operation of the NPP is performed using established industry and safety standards, and that the NPP is managed and operated by well-trained and qualified personnel following laid-down safety guidelines and procedures. As the operation of NPPs can pose radiation threats, the public and the environment have to be protected against these threats. Governments ensure this protection by only allowing the operation of NPPs under formal licenses.

For the purpose of licensing, the RB conducts a thorough review of all the phases in the life of the nuclear power plant. This may start with a formal appraisal of the technology to be deployed, continuing with an assessment of the site characteristics and the design and engineered safety features of the selected NPP, as well as specification of the safe operating envelope and other licensing conditions for operation of the NPP. The RB also maintains a careful oversight during the entire operational phase of the NPP by reviewing periodic reports, by making regulatory inspections and by other means to ensure that the licensing conditions are being complied with on a continuing basis. At the end of the plant's operating life, the licensee ensures that the NPP is maintained in a safe state until its complete dismantling and decommissioning is taken up in accordance with the stipulations made by the RB. Finally, the RB reviews the decommissioning plan for the NPP and authorizes it if the requisite safety criteria are met, including those for disposal of radioactive waste arising from decommissioning activities.

20.2.1 General considerations

When licensing a nuclear power plant, some general safety considerations as well as detailed requirements have to be taken into account. Some of the most relevant general considerations are:

- The plant and the site on which it will be built are closely related and have a mutual interaction. There should not be any unacceptable adverse impact from plant operation on the site and, similarly, no unacceptable adverse impacts from site characteristics on the safety of the plant.
- There is assurance of control of reactivity, reactor core cooling and containment of radioactivity; these three basic safety functions have to be achieved at all times, under all design basis conditions including design basis accidents. For beyond design basis accident conditions, it should also be possible to control the progression of an accident and mitigate its consequences.
- There is a close relationship between the safety of the NPP and the persons operating it, i.e., the human–machine interface. It is therefore important that the plant is operated by well-trained and qualified personnel to ensure that the plant operating configuration and its process parameters are kept within the safety envelope and license conditions prescribed by the RB.
- Security measures and emergency preparedness plans should be in place and tested satisfactorily before nuclear fuel is loaded in the core.

Various other licensing requirements should be clearly prescribed by the RB for each one of the phases in the life of the plant. When a license is given in sequential steps, each step normally includes an explanation of the basic requirements for the following step. Some of these requirements include the following:

- The regulatory process for the various stages of licensing of the NPP should be clearly laid down by the RB in a formal manner that should include a list of technical documents to be submitted by the applicant, the lead time for their submission, the list of safety requirements and standards to comply with, and the methodology for their detailed review within the RB.
- The Site Evaluation Report (SER), the Preliminary Safety Analysis Report (PSAR) and the Final Safety Analysis Report (FSAR) are the primary documents submitted by the applicant to the RB in support of the site, construction and operating license applications, respectively. These reports and their supporting technical documents should meet the RB's specifications and should be of a high quality and in sufficient detail.

- The RB should carry out inspections during manufacture of safety-related components to confirm that they meet the prescribed standards. Likewise, the RB will conduct periodic inspections of the NPP during its construction phase to ensure that the construction of the safety-related systems, structures and components (SSC) meets the safety and quality standards.
- On completion of construction, management of the NPP is transferred from a construction group to a commissioning and operations group. The licensee submits an application to the RB for authorization of commissioning activities, according to a well-defined sequence and detailed procedures for all activities. After a satisfactory review, the RB authorizes commissioning. Initial fuel loading in the reactor core marks the start of operations and hence needs authorization from the RB. At this stage, a complete operational discipline must be in force with a full complement of trained and authorized operational personnel in position, along with security and emergency plans satisfactorily tested and in place. Subsequently, the RB authorizes the raising of the reactor power in predefined steps, each step being reviewed as appropriate.
- During the operational phase of the NPP, the RB reviews periodic operational reports, accounts on safety-related incidents and ageing status of the SSCs to confirm that the NPP continues to successfully meet the applicable license conditions and current safety standards.
- At the end of its operating life, the NPP is decommissioned, though only after the RB issues a license for this purpose after a review of the decommissioning plan.

20.2.2 Licensing stages

Each phase in the life of a nuclear power plant requires a license or approval from the RB. Table 20.1, derived from Annex 1 in INSAG-22, describes the main phases and the safety infrastructure needed by countries to establish and maintain a licensing process (INSAG, 2008a). The major phases and the corresponding licensing stages for an NPP are the site, construction, commissioning, operation and decommissioning licenses. Some of these licenses may be divided in sub-stages like ground breaking, first pour of concrete, and the erection of major equipment to facilitate working out the detailed design in parallel with the civil construction work. In that case, the requirements of safety review and submission of technical documents for each sub-stage, together with their submission schedule, should be clearly specified. Conversely, it may be decided to issue the construction and operating licenses in one step, in which case the entire design, including its details, should be submitted before the start of the review process.

Table 20.1 Major activities required by applicants and regulatory bodies at each of the main phases in the life of a nuclear power plant

Phase	Duration (years)	Applicant activities	Regulatory activities
1. Design certification	3–4	• Submit documents in accordance with existing regulations and guidance.	• Develop requirements and guidance. • Conduct public participation.
2. Siting	2–3	• Select and characterize the site: seismicity, extreme meteorology, flooding, man-made external events, combinations of the above, population density, use of affected land. • Demonstrate compatibility between site and plant. • Submit documents in accordance with requirements.	• Develop requirements for siting. • Evaluate compatibility of site parameters with plant design. • Evaluate compatibility of the plant with the site. • Evaluate compatibility of site and plant with emergency planning.
3. Design and construction	5–7	• Develop a preliminary safety analysis report and other required documents. • Construct the plant in accordance with the project and the terms in the license. • Assure quality during design, procurement, assembling and testing.	• Develop requirements for design, construction, testing and quality assurance. • Establish a programme of inspections covering equipment fabrication, plant construction, testing and quality assurance.
4. Commissioning	1–2	• Establish an organization to transfer the plant from construction to operation. • Develop a plan for testing and plant acceptance. • Develop the final safety report and apply for operating license. • Prepare operators for accreditation.	• Develop requirements for commissioning and operation. • Create a body of experts to witness tests and accept their results. • Establish a process for license reactor operators.

Table 20.1 Continued

Phase	Duration (years)	Applicant activities	Regulatory activities
5. Operation	40–60	• Operate and maintain the plant in accordance with established requirements. • Perform periodic testing and inspection on components, systems and structures relevant to safety. • Conduct self-assessments and external peer reviews. • Retrofit operating experience. • Conduct emergency drills.	• Develop requirements for operation, maintenance, testing, reporting events, conduct self-evaluations, peer reviews and feedback from operating experience. • Create a continuous oversight system and a body of resident inspectors, enlarged inspection teams and safety evaluators.
6. Plant modification	1–2	• Define the need and cause for modifications. • Analyze the impact of the modifications on the overall safety of the plant. • Submit the application for approval. • Conduct the modification in agreement with the applicable standards and conditions.	• Determine and request the need for plant improvements. • Be prepared to evaluate plant modifications by establishing guidance for submittals and procedures for evaluation. • Verify that improvements and modifications comply with the applicable standards and conditions.
7. Operation renewal	2–3	• Develop documentation for submittal of a request. • Introduce an ageing management system. • Review the safety analysis report and the probabilistic safety assessment to prove the validity of the renewal.	• Develop requirements for operation renewal. • Develop capacity to evaluate the ageing phenomena and their impact on the safety of the plant. • Develop an inspection and oversight programme to verify the progress of ageing.
8. Decommissioning	5–10	• Decide and prepare for the shutdown of the plant. • Develop and implement a plan for decommissioning and radioactive waste management in accordance with the applicable requirements.	• Develop requirements for decommissioning. • Evaluate the decommissioning plan and the plan to manage radioactive wastes. • Create a system to inspect and monitor dismantling operations.

Source: derived from Annex 1 in INSAG-22 (INSAG, 2008a).

An operating license is normally issued for the design life of the NPP. However, during the long operation period, which may extend to several decades, the safety status of the NPP is reviewed from time to time, for example by conducting detailed periodic safety reviews. This is to confirm that the NPP, in spite of the ageing of its structures, systems and components (SSCs), meets the current safety requirements and is likely to continue to do so until the next safety review. Towards the end of the license period, if requested by the operating organization, the operating license may be extended for a further period provided a detailed safety review clearly establishes that the NPP can be operated safely for that length of time.

After the NPP is finally shut down at the expiry of the operating license, or due to economic or other reasons, there is likely to be a waiting period to allow for the natural decay of short-lived radionuclides, to reduce the radiation fields on the SSCs to make their dismantling, handling, packaging and transportation to a radioactive waste disposal site easier. Dismantling should never start while fuel is still in the reactor core or in the used fuel decay pool, since as long as nuclear fuel is present an NPP is considered operational and the relevant licensing conditions continue to apply. Even after the fuel is removed from the core and the decay pool, an NPP will have to be kept under surveillance to ensure that there is no undue exposure of plant personnel to radiation, and that no unauthorized release of radioactivity is made to the environment. The NPP's license should be modified appropriately during such periods.

20.2.3 Licensing models

A large variety of examples of licensing methodologies can be found in updates published by the Nuclear Energy Agency (NEA) on analytical studies of nuclear legislation in Member States (NEA, 2004). Likewise, Chapter 2 of the IAEA *Handbook of Nuclear Law* (Stoiber *et al.*, 2010) recognizes the large variety of regulatory organizations. The *Handbook* also includes recommendations on the establishment of such bodies. For illustration, three very different and relevant examples are discussed in the Appendix in Section 20.10.

20.3 Licensing application and supporting technical documents

The licensee is required to prepare an extensive set of documentation covering all aspects of the plant's life cycle. Some of these documents are mandated by national laws and regulations, while others are required by the regulatory body for licensing or in response to specific requests. The content and number of documents will vary considerably from country to country depending on the national legal and regulatory systems and practices. However, the IAEA has developed recommendations on the documentation requirements that provide some common guidance, while recognizing that other systems may also be effective (IAEA, 2002a). Whatever system is used, the IAEA Guide states that:

> The system of regulations should provide advance information to the operator on the requirements for each major stage of authorization. This will assist the operator to make sound plans and decisions with respect to safety in the siting, design, construction, commissioning, operation and decommissioning or closure of a nuclear facility.

While it is not possible to cover the specific documentation requirements that apply to all countries, Table 20.2 contains a representative sample of the types of information that might be required at each stage.

20.3.1 Choice and characterization of a site

Many potential sites can be considered for an NPP as long as all the site characteristics that could impact safety are understood and can be addressed for that location; similarly, the plant characteristics should not unduly

Table 20.2 Examples of information that is required for the licensing of a nuclear power plant

License	Representative information
Site preparation	– Site description, activity to be performed, exclusion zones, structures, location – Site characteristics, meteorology, seismology, geology and other natural phenomena – Site evaluation process and investigations – Preparatory work on site and surrounding area – Programme for determining the site's environmental baseline – General characteristics of the plant that would affect the site – Effects on environment, health and safety, and measures to mitigate these effects
Construction	– Preliminary Safety Analysis Report (PSAR) – Other documents and detailed analyses supporting the PSAR – Other documents covering topics that may be excluded from the PSAR as required by national practice; examples could include compliance with security measures and safeguards – Assessments that are required for regulatory agencies other than the nuclear regulatory body; an example could be a detailed environmental assessment

Table 20.2 Continued

License	Representative information
Operation	– Final Safety Analysis Report (FSAR); the FSAR would expand on the PSAR, including results of the pre-nuclear commissioning and future commissioning plans, changes to the as-built design basis, and status of training and recruitment for operations – Other documents and detailed analyses supporting the FSAR, including any updates to the existing PSAR supporting documents – Other documents covering topics that may be excluded from the FSAR as required by national practice; examples could include compliance with security measures and safeguards – Assessments that are required for regulatory agencies other than the nuclear regulatory body
Decommissioning	– Description and schedule – Planned state of site on completion of decommissioning – Description of nuclear substances, hazardous substances, land, buildings, structures, systems and equipment affected – Measures, methods, and procedures for carrying out decommissioning – Disposition of used nuclear fuel – Transportation of nuclear and hazardous materials – Compliance with safeguards and national security – Nature and extent of any radioactive contamination at facility – Effects on environment, health and safety, and the measures to mitigate the effects – Location of release points, maximum quantities, and concentrations, volume and flow rate of nuclear and hazardous substances, and measures to control the releases – Measures to prevent or mitigate the effects of accidental releases and the emergency response plan – Qualification requirements and training programmes

impact the site. Site evaluation and licensing for site preparation by the regulator may also require that some general characteristics of the NPP be specified, such as the power generated and resource requirements (such as cooling water). The licensee should either specify the site characteristics in the bid documentation or provide an envelope of conditions into which the various candidate sites fall. This means that the site criteria must be identified early in the deployment of an NPP project. (Chapter 18 considers how nuclear sites should be selected and characterized.)

Once a site has been identified as a potential host for a nuclear plant, it must be evaluated. The IAEA has published a Safety Requirements document for site evaluation that establishes the requirements for the various elements of a site evaluation (IAEA, 2003a). The publication deals mainly with site-related factors that could impact on severe events of low probability to ensure that the site–installation combination does not constitute an unacceptable risk over the lifetime of the nuclear plant.

There are three important safety considerations when evaluating a potential site, as stated in the IAEA Safety Requirements (IAEA, 2003a). The first of these is an evaluation of the effects of external events on the safety of the plant. These evaluations cover both external human-induced events and natural phenomena. The second consideration is an evaluation of the nature of the site and its environment that could have an impact on the exposure of people and the environment to radioactive material. The third concerns the characteristics of the areas external to the NPP site that could impact on the implementation of emergency procedures; these include such characteristics as population density and transportation routes. If there are any deficiencies arising from the evaluation of these considerations that are not addressed by the design of the plant, by the measures for site protection, or by administrative procedures, then the site cannot be licensed for a NPP. To assist with site evaluation, the IAEA has published a number of Safety Guides for each of the important site factors that need to be considered: seismic design (IAEA, 2003b), external human-induced events (IAEA, 2002b), dispersion of radioactive material (IAEA, 2002c), meteorological events (IAEA, 2003c), flood assessment (IAEA, 2004a), and geotechnical aspects (IAEA, 2005). The 2011 Fukushima Dai-ichi event in Japan clearly indicates the need to consider the existence of two or more natural and even man-made events, such as an earthquake followed by a tsunami in coastal sites, or a high tide coinciding with a rain flood in estuary sites, or nearby explosions releasing highly toxic chemicals, which may affect the plant and the emergency activities of operating personnel. Nuclear emergency planning should also consider the possibility of a nuclear emergency occurring concurrently with a general emergency produced by a natural or man-made cause.

Using the IAEA criteria, it is possible to establish a logical site selection process. First, a number of sites or regions could be selected based on national priorities. The licensee would then perform a screening assessment that would eliminate any sites that do not meet the licensing criteria. The screening process would usually take into account any site characteristics that could impact safety. These characteristics include population distributions around the site, meteorological conditions, seismic phenomena, volcanic activity, flooding, soil types, groundwater characteristics, and other geotechnical and hydrological considerations. Next, the remaining sites would be verified according to a set of predefined site exclusion criteria. This could then be followed by confirmation of the results of the previous steps through site investigations and laboratory measurements, along with the preliminary plant characteristics such as loads, physical dimensions, and preferred layouts. The licensing submission would include all such information and would be subject to independent assessments by the RB.

The initial site evaluation process also establishes the basis for longer-term requirements that will remain in place throughout the lifetime of the plant. The pre-operational phase includes ongoing assessment work during construction to refine the characterization of the site. During the operational phase, continuous monitoring and assessment of site characteristics will be required as part of the operating license. Also, if there are any significant changes in population distributions or human activities surrounding the plant, or a change to the nuclear capacity on the site, these changes will have to be taken into consideration.

20.3.2 The selected design and its safety review

The design selected must undergo a formal design safety review process by the regulator before a construction license can be issued. This review determines whether the design of the selected technology meets the required national safety regulations. Normally, these regulations will be consistent with the IAEA Safety Standards, which constitute the international consensus on nuclear safety in the form of Principles, Requirements, and Guides. A recent overview of the current status of the relevant IAEA documentation is available from the IAEA (IAEA, 2010a). Chapter 9 discusses current and near-future available technologies.

The IAEA has published a Safety Requirements document to establish the generally applicable requirements for a safety assessment of nuclear facilities and activities (IAEA, 2009a). The major licensing document the licensee must provide for a construction license is the PSAR for the selected NPP design. This is a comprehensive document running to thousands of pages of technical information, backed up by detailed analyses, R&D results, and other supporting documentation. Table 20.3 lists the content and some examples of the scope of material that needs to be covered (IAEA, 2004b). It is clear from Table 20.3 that the licensee must be familiar with all aspects of the NPP life cycle: design, construction, commissioning, operations, and decommissioning. The PSAR is the licensee's evaluation of the safety basis for the plant covering its entire plant life cycle.

The production of the PSAR is a major task. Normally, the vendor/designer provides much of the non-site specific technical information for this document. The licensee must provide the information that is specific to the country building the NPP, such as site conditions and operating

Table 20.3 Safety analysis report content

SAR chapter	Chapter scope and examples
Introduction	This chapter deals with general issues that are country and project-specific.
General plant description	Topics include a description of the applicable codes and standards, the basic technical characteristics of the technology, the plant layout, plant operating modes, and the documents and analyses incorporated by reference.
Management of safety	Specific aspects of management processes are described along with the monitoring and review of safety performance.
Site evaluation	This includes site reference data such as hydrology, meteorology and seismology, as well as the evaluation of site-specific hazards and activities at the site that could influence the plant's safety. The proximity of industrial, transport and military facilities is also described. Site-related issues for emergency planning and accident management are developed. Also, monitoring of site-related parameters and a description of radiological conditions are included.
General design aspects	The safety objectives and design principles, and conformance with the design principles are discussed. The classification of structures, systems, and components is also addressed. Specific topics include civil engineering works and structures, equipment qualification, environmental factors, human factors engineering, and protection against internal and external hazards.
Description and conformance to the design of plant systems	This is a comprehensive discussion of the reactor components. These include the reactor coolant and associated systems, the engineered safety features, instrumentation and control, electrical systems, plant auxiliary systems, power conversion systems, fire protection systems, fuel handling and storage systems, radioactive waste treatment systems, and other safety-related systems.
Safety analysis	Acceptance criteria for the safety objectives are stated. A summary of the results of the safety analyses to meet the acceptance criteria is presented.
Commissioning	This describes how the various SSCs will be tested and verified to meet the design requirements.

Table 20.3 Continued

SAR chapter	Chapter scope and examples
Operational aspects	Operations includes a large range of topics: the organization, administrative procedures, operating procedures, emergency operating procedures, guidelines for accident management, maintenance, surveillance, inspection and testing, core management and fuel handling, management of ageing, control of modifications, qualification and training of personnel, human factors, programme for operational experience feedback, documents and records, and outage management.
Operational limits and conditions	This defines the safe operating envelope for the plant.
Radiation protection	The application of the ALARA principle is discussed. Radiation sources, design features for radiation protection, radiation monitoring, and a radiation protection programme are included.
Emergency preparedness	This addresses emergency management, emergency response facilities, and capability for the assessment of accident progression, radioactive releases, and the consequences of accidents.
Environmental aspects	Both radiological and non-radiological impacts of the NPP are discussed.
Radioactive waste management	Topics include the control, handling, minimizing, handling, conditioning, storage, and disposal of radioactive waste.
Decommissioning and end of life aspects	The decommissioning concept for the NPP is presented, and includes provisions for safety, the differing approaches to decommissioning, and planning of the preliminary work.

Source: IAEA (2004b).

organization information. Notwithstanding the contributions from the vendor/designer, the licensee must have access to expertise for each of the PSAR areas. It is not possible to operate an NPP safely without this knowledge, whether available internally or obtained externally through technical support organizations. The latter would include access to R&D facilities capable of handling and characterizing radioactive components.

The regulator performs a detailed independent assessment of the PSAR and usually presents its results in a safety evaluation report (SER). The SER then becomes the technical basis for awarding or denying the construction license and for establishing the required limits and conditions to be complied by the licensee. Therefore, the licensee must be prepared to respond effectively during the evaluation process to any detailed technical questions from the regulator on the PSAR topics using internal or external expertise.

20.3.3 Regulatory documents for commissioning and operation

The overall objective of commissioning is to prepare the SSCs for operation. This involves verifying that the SSCs meet their design requirements for safety and performance, for both individual structures and components and integrated systems. These requirements cover normal operation, anticipated operational occurrences, and design basis accidents. Verifying the design provisions for management of accidents beyond the design basis can also be done at this stage, as far as it is feasible. There is some overlap between construction and commissioning since some SSCs may be commissioned before completion of the entire plant. (The various aspects of commissioning and related activities are considered at length in Chapter 22).

There are several steps during commissioning that may require regulatory approval. The introduction of fissile material into the plant is an important event and is considered in some cases to be the first point where regulatory decisions are required. Since commissioning is performed typically over a few months, the licensee and the RB must both be prepared for an intensive period of activity. Besides planning and organizing its own activities, the licensee should ensure that the RB establishes and communicates a detailed plan outlining how it will review the commissioning work, the nature of the required approvals and hold-points, and what information is required to be submitted by the licensee at each hold point. For example, the licensee should understand the clearances that the on-site regulatory staff can issue at the various stages of commissioning, and the submissions that are required to ensure such clearances. The licensee must also be sensitive to the fact that results of commissioning could lead to further refining of the regulatory requirements for plant operation, for example in its operating procedures and in-service inspections requirements.

An operating license requires the submission of FSAR based on the PSAR previously submitted for the construction license, as summarized in Table 20.3. However, it includes more information from both the construction and commissioning programmes and may also be impacted by new R&D information and international safety developments that have arisen during the construction period. Obviously, the satisfactory completion of the training and certification of operating staff is an essential milestone for the operating license, and is considered further in Section 20.5.4.

Operational procedures are developed before a plant is transferred from construction to operations. These include procedures that cover normal and off-normal operations, surveillance, maintenance, and emergency operations. Emergency operations procedures normally have to be approved by the regulator before issuing the operating license and prior to initial fuel loading. Several other submissions could be required depending on the national licensing processes and FSAR content, as indicated in Table 20.3.

During operation, there will be ongoing requirements to submit various operational reports to the regulator depending on licensing requirements and on the occurrence of any events that impact or have the potential to impact safety. Some of these requirements are discussed in Section 20.5.

20.3.4 End of life and requests for decommissioning

Decommissioning begins to be addressed at an early stage of a nuclear power plant programme. As noted in Table 20.3, the Safety Analysis Report includes a decommissioning concept including provisions for safety, the differing approaches to decommissioning, and planning of work. The end state for decommissioning, depending on national legal and regulatory requirements, encompasses partial or full decontamination and/or dismantlement, with or without restrictions on further use of the site. The IAEA has developed basic safety requirements that must be satisfied during the planning and implementation of decommissioning, for the termination of practices and for the release of facilities from regulatory control (IAEA, 2006a). Chapter 24 describes the various aspects of decommissioning and the experience already gained.

There are three general approaches that could be followed to achieve a decommissioning end state. In all three cases, a facility is eventually released for other uses, either with or without regulatory restrictions, but the time frames are different. The first approach is immediate dismantlement, where radioactive contaminants are removed or reduced to a level that permits the facility to be released. For this approach, the decommissioning project would need to be initiated shortly after the end of plant operations. It requires timely completion of the decommissioning site activity and removal of radioactive material from the NPP to a licensed facility, followed by processing for either long-term storage or disposal. The second approach is deferred dismantling or safe storage. In this case, any SSCs that have radioactive contaminants are either processed or placed in a condition where they can be safely stored and maintained. Subsequently, the SSCs are decontaminated and/or dismantled such that the facility's radioactivity returns to levels that allow the facility to be released. The third approach is entombment. For this approach, the radioactive SSCs are safely encased until the radioactivity decays to a level such that the facility can be released from regulatory control.

Whatever approach is taken, the licensee must ultimately develop a final decommissioning plan for regulatory approval. The development of this information will likely require a preliminary period of work before the decommissioning plan can be finalized and be submitted to the regulator. The plan might encompass the strategy, the current state of the plant including radiological characteristics, the schedule, implementation and management of the plan, how the waste will be managed, and a description of the end state and how it will be verified.

The licensing submission will also require a safety assessment that may include some of the topics in Table 20.2. The assessment would cover the decommissioning activities given in the plan and any potential abnormal events that could occur. The occupational exposures and the potential releases to the environment, and the health and safety of the public, would be addressed, including the mitigation and prevention strategies. The IAEA recommendations for the development and review of the decommissioning safety assessments are given in a Safety Guide (IAEA, 2009b), where it is stated:

> Decommissioning activities are performed with an optimized approach to achieving a progressive and systematic reduction in radiological hazards, and are undertaken on the basis of planning and assessment to ensure the safety of workers and the public and protection of the environment, both during and after decommissioning operations.

The site can be released from regulatory control once the licensee has completed the decommissioning work and has met the regulatory requirements. Recommendations for meeting these requirements are the subject of an IAEA Safety Guide (IAEA, 2006b). The Guide is directed to both the regulatory body and the licensee, and covers the release of sites or parts of sites from regulatory control after a practice has been terminated.

20.4 Safety review of licensing applications and license requirements

License applications are formally submitted to the RB. The RB first verifies that the information provided in the application is sufficient for conducting a proper safety review; if this is not the case, the applicant is requested to submit the information required. Once the RB is satisfied, the application is formally accepted and a schedule for the evaluation process is established. The way in which the evaluation is conducted is country dependent but there should always be a person managing the process and a large group of specialists for the various fields of experience required. The end point of the evaluation is the completion of an SER.

The RB may not be sufficiently knowledgeable in certain specific areas; in such cases, help can be obtained from various sources. For new entrants, the most relevant source would in most cases be the RB in the country of origin of the NPP design, though an experienced group of international senior regulators could also help establish a knowledge base in a new regulatory system. The IAEA can provide a variety of services, including advice to the assessment manager. Technical help and advice can also be requested from Technical Support Organizations (TSOs), as is very common in the European practice, available in the country's own institutions, such as nuclear research organizations, universities and academies. Contracts with national and international private institutions could also support the evaluation process. This help does not take away from the RB the responsibility of preparing the SER which results from such analysis.

The SER is a substantial document, generally developed by following well-established procedures. The main aim is to verify compliance with the regulatory requirements applicable to the license under consideration. During this process, generally large lists of clarifying questions or requirements for further information are addressed to the applicant. A prompt and precise response to the RB enquiries speeds up the evaluation process. The SER ends with a final evaluation and with a complete set of limits and conditions to be followed when performing the activities envisioned in the requested license. It may also include a reference to the requirements for the next license.

20.4.1 Site license

As already described, an application for a site license, whether part of a more general license or a stand-alone application, needs to identify the precise site on which the applicant proposes to build a nuclear power station, and the characteristics of the site need to be described, as do the mutual interactions between the plant and the site. The submitted documentation should be analyzed by the RB against established safety principles, such as:

- *The expected frequency and magnitude of external events affecting the safety of the NPP.* The RB has to verify the validity of the data collected and the magnitude of the expected external natural and man-made hazards, including seismic disturbances, extreme weather events such as flooding, nearby explosions, large releases of chemical contaminants or extended fires. The Fukushima 2011 accident and other experiences have shown that there could be possible combinations of related natural phenomena and human-induced events.
- *The requirements for an efficient emergency plan*. Emergency planning is the last resource at hand to mitigate the consequences of serious accidents in nuclear power plants. The proximity of schools, hospitals and other welfare institutions will feature in considering the feasibility

of implementing emergency countermeasures (including possible evacuation of areas around the site). As already mentioned, the potential difficulties of a nuclear emergency within a more general emergency should also be considered by the RB.
- *The social, economic and environmental effects of the NPP*. An NPP will have effects on the surrounding population and the environment. The RB should consider the population density and the proximity to large and medium cities, technological parks, recreational areas, national parks and heritage locations which may become heavily affected by radiation releases of a certain magnitude.

The analysis by the RB experts requires knowledge and experience in earth sciences to determine the magnitude of the maximum possible natural events, as well as experts on man-made events, and people with experience of emergency planning. The best help to the RB will probably come from national institutions dealing with such phenomena and activities. A site license will typically contain requirements and limitations on the site's preparation activities that may be conducted before construction begins.

Many RBs require that a local environmental impact statement (EIS), assessing the impact caused by the future power plant, is also submitted by the applicant and analyzed by the RB at the time of site evaluation. It is also customary at this stage to inform stakeholders of the project and to allow them to formally make representations detailing any reservations they may have about the project. Some countries, for example the UK, require that a Public Inquiry should be called at which the applicant is invited to present their case and to hear and consider the contributions and concerns raised by participating stakeholders. The involvement of stakeholders in nuclear issues before deciding the construction of a new nuclear power plant is recommended in INSAG-22 (INSAG, 2008a).

20.4.2 Construction license

The construction license requires the submittal of a PSAR, already described in Section 20.3.3. As part of the submission, the licensee has to provide details of the project management arrangements and quality assurance provisions. The RB seeks assurance that the work will be conducted safely and in accordance with the environmental and transportation requirements of the terms of the license, and that the installation conforms to the approved design.

The safety assessment of this documentation requires a major display of the RB's resources, and arrangements for obtaining external help and advice may need to be made. The analysis covers all the chapters of the PSAR and related information, conducted in accordance with pre-stated procedures. All evidence provided by the operator in support of the request for a construction license needs to be checked and verified by the RB, partially by scrutinizing the operator's analyses but also often by performing independent analyses. The SER for the construction license will form the basis for the content of the license, its limits and conditions.

The analysis of the chapter on potential accidents requires expertise, as it is necessary to verify that potential accidents can be avoided or controlled. A deterministic approach is generally used: a set of potential accident scenarios is proposed, which the NPP design includes equipment and procedures to manage. This constitutes what is called the design basis. A new methodology, a Probabilistic Safety Assessment (PSA), has started to be used as a complement to the deterministic approach. Recently, INSAG recommended integrating both approaches (INSAG, 2011).

Throughout construction, it should be ensured that, once approved, no alteration or amendment is made to the plant and equipment, or to any approved arrangement, unless the RB has approved such alteration or amendment. Normally construction schedules are divided into installation stages. The RB can specify hold points, beyond which work may not progress without its consent. Throughout construction, the RB carries out a programme of inspections, assessments and reviews of the activities performed. If at any stage the RB is not satisfied, a variety of options should be put into practice to improve the situation, including stopping all work until the issues in question are addressed. To achieve a high quality of systems and components relevant to safety, the components need to be qualified to properly respond to seismic and extreme environmental situations. As far as possible, components that have already been proven in operation should be used. Manufacturing must conform to high quality standards.

The construction phase is considered complete when SSCs relevant to safety are tested under well-defined conditions and established standards. Examples of these pre-nuclear tests include pressure tests of the primary coolant system (including the reactor vessel), performance tests of emergency coolant systems, containment pressure and leakage rate tests, and electrical systems performance tests. Representatives of the RB, or specialists working on their behalf, generally witness these tests for acceptance by the RB.

20.4.3 Commissioning license

The construction license establishes the prerequisites under which the commissioning of a plant is conducted. These conditions include arrangements under which the RB will specify what is to be done, conditions under which approval must be given and where consent has to be sought before moving to the next stage of commissioning. Chapter 22 covers the activities to be conducted during commissioning.

Well before the end of construction, the licensee makes and implements adequate arrangements for commissioning the NPP, mainly with regard to those processes that may affect safety. Such arrangements need the approval of the RB. These arrangements provide documentation to justify the safety of the proposed commissioning. It is recommended that the licensee appoints a suitably qualified person or persons to control, witness, record and assess the results of any tests carried out in accordance with the requirements of the commissioning arrangements. The licensee should ensure that full and accurate records are kept of the result of every test and operation carried out in pursuance of these requirements.

The RB considers the proposed arrangements for commissioning very closely and makes preparations for licensing this phase, and for witnessing the nuclear tests that are to be performed and analyzed. The RB creates a commissioning group with experience on the subject but, if that experience is missing, outsiders can be hired (usually commissioning experts from the RB in the country of origin of the project). Nevertheless, responsibility for making an analysis of the commissioning programme, establishing the limits and conditions for their conduct, witnessing the tests and accepting the results remains with the RB.

It is customary to divide the commissioning into stages. If the RB so specifies, the licensee should not commence commissioning, or proceed from one stage to the next, without the consent of the RB. The licensee should ensure that, once approved, no alteration or amendment is made to the arrangements unless the RB agrees to such alteration or amendment. The commissioning phase is terminated when all foreseen tests have been successfully conducted and approved by the RB.

20.4.4 Operating license

In many regulatory organizations the commissioning phase just described is only one part of the operating license. In any case, the licensee has to submit an FSAR, together with a series of additional documents relevant for the operational phase, as described in Table 20.3. The FSAR is a refinement of the PSAR, describing the NPP as it has been built and introducing changes in equipment capabilities and operational limits as determined by the results from the pre-nuclear and nuclear tests undertaken. Apart from organizational documents, one of the most significant documents for operation is the Operational Limits and Conditions, also called the Technical Specifications for Operation.

These documents are reviewed by the RB and form the basis of the operating license. By this time, even for new entrant countries, the RB staff will already have accumulated a great deal of experience and expertise; nevertheless, when confronted for the first time with handling the operating license, outside help will be needed. As in previous cases, it is recommended that advice be sought from the RB in the country of origin of the project, from the IAEA, or from TSOs experienced in operating research reactors. In the longer term, the RB will become more independent from external help, except in cases regarding anomalous situations and accidents. In any case, the RB will need to convene a competent body of experts, including those who have been involved in the construction and commissioning phases. Once again, the SER constitutes the basis for the operating license and its limits and conditions.

The operating license covers a great variety of different issues in some detail. Among other things, the license stipulates the procedures according to which an activity is to be carried out, the conditions to be respected, the documentation the operator has to produce, what they have to report to the RB, and whether the participation of a representative of the authority is required.

The license therefore quotes the document on personnel organization which describes the functions, responsibilities and tasks of persons and organizational units. It states the requirements for training for important positions. Operating activities are undertaken according to a number of internal regulations. Those with safety relevance are licensed. Important among these internal regulations are manuals on in-service testing, maintenance, radiological testing, shift and control room organization, access and security, alarms, physical protection, and quality assurance.

Operation of the plant is guided by procedures for normal operation and for incidents and accidents. Limits and conditions for operation have to be respected. There are also guidelines and procedures for severe accidents, and these documents also form part of the license. There are requirements on the information the authority needs for fuel outages, on the justification of the safety of the new core, and the conditions for restart. The procedure for how plant modifications are to be processed is also fixed at this stage. Additionally, there are requirements on quality assurance for components to be exchanged, with special regard to core internals.

The license stipulates that the operator has to follow and analyze incidents in other plants, and justify their conclusions regarding their own plant. It also states how the operator should proceed with reportable events in the plant itself. The license deals with the proof of waste disposal, and the handling of fuel and radioactive waste. Of course, it sets limits for tolerable effluents in air and water. The license is also a basis for the surveillance which the authority will perform during operation. Therefore, it regulates the documentation which the operator has to maintain and the reports they must submit to the authority on a regular basis.

Regular inspections, either announced or not, are conducted by the RB. Some RBs have resident inspectors assigned to each NPP who oversee day-to-day operations and report to the RB headquarters. In case of anomalous situations, a so-called reactive inspection is put into effect to analyze the situation and oversee the actions taken. Some RBs have established permanent oversight systems, based on selected safety pillars; non-compliance with the defined pillars generates a colour code which measures the importance of the non-compliance, which is maintained until the non-compliance is addressed and corrected.

20.4.5 Decommissioning license

An NPP will cease to be operational if a decision is taken to retire it from service, at the end of its licensed operating life or earlier. Several causes may dictate the earlier termination of an operating NPP, for example a decision by the licensee for economic or other reasons, the cancellation of the operating license by the RB, or the impossibility of recovery from an accident. The licensee should formally communicate to the RB about such a decision and the proposed arrangements for safekeeping of the facility pending its decommissioning.

The RB will review the proposal and appropriately modify the operating license. This includes changes in the technical specifications for operation and other licensing conditions, like those related to requirements of operating staff, in-service inspection, and surveillance and operability of equipment to maintain the facility in a safe state. However, as long as nuclear fuel is present in the reactor core, the NPP is considered operable and the complete operational discipline should remain in force. After the reactor core is completely defuelled, the operating license may be terminated. However, the facility will still remain under regulatory control, with appropriate safety requirements specified, as long as radioactive material is present at the site, pending its final decommissioning.

Most RBs have enacted regulations on decommissioning commercial nuclear power plants, with these regulations covering the time from termination of operation to when the site is declared fit for unrestricted use. In any case, the licensee declares that the reactor has been shut down permanently and that they are ready to request a decommissioning license. The licensee keeps its prime responsibility as long as there is fuel on the reactor premises, either in the reactor core or in the spent fuel decay pool. After removing the fuel, responsibility for the site can be transferred to the agency conducting the dismantling.

The operator performing the dismantling should submit a safety analysis report to the RB describing the decommissioning activities to be conducted and the safety provisions that have been made to comply with the existing regulations. Attention is given to decontamination activities, to radiological protection of workers and the environment, and to the management of radioactive waste. The RB evaluates the information received and prepares an SER with the proposed limits and conditions to be complied with during the process, these mostly being the acceptable residual radioactivity level remaining on the site for it to be released for unrestricted use, the reports to be submitted on the conduct of operations, and hold points for inspection. At the end of the process, a radiological survey of the site is generally conducted before releasing and declaring that decommissioning has ended.

20.5 Licensee activities during design, construction, commissioning, operation and decommissioning

The differing roles of the operator and of the regulator with respect to safety need to be made clear. The IAEA's *Fundamental Safety Principles* require that 'the prime responsibility for safety must rest with the person or organization responsible for facilities and activities that give rise to radiation risk' (IAEA, 2006c). That is, the licensee is responsible for safety, and the regulator is responsible for granting licenses and providing oversight of the operator's activities with respect to safety. These responsibilities persist throughout the entire life cycle of an NPP and can span a period of several decades. The license specifically states that the licensee is responsible for the following:

- Establishing and maintaining the necessary competences
- Providing adequate training and information
- Establishing procedures and arrangements to maintain safety under all conditions
- Verifying appropriate design, and the adequate quality of facilities and activities and of their associated equipment
- Ensuring the safe control of all radioactive material that is used, produced, stored or transported
- Ensuring the safe control of all radioactive waste that is generated.

On leadership and management for safety, Principle 3 of the *Fundamental Safety Principles* states that 'Leadership in safety matters has to be demonstrated at the highest levels in an organization'. Therefore, the starting point for a licensee is senior management's leadership of, and commitment to, safety through a clearly articulated safety vision that is communicated to every employee. Or, as the International Safety Group (INSAG) expresses it (INSAG, 2002):

> Commitment to safety and to the strengthening of safety culture at the top of an organization is the first and vital ingredient in achieving excellent safety

performance. This means that safety (and particularly nuclear safety) is put clearly and unequivocally in first place in requirements from the top of the organization, and there is absolute clarity about the organization's safety philosophy.

The next step is to ensure that a safety culture, leadership and management systems and processes are all in place to ensure safety. These must be established early in the NPP project and certainly before the bidding process begins. For a new NPP operator, assistance from an experienced operator of a similarly designed NPP is likely to be essential for establishing these requirements. Once established, the licensee must communicate its safety policies on an ongoing basis to both staff and its suppliers. For example, bid specifications should clearly reflect the operator's safety requirements. Also, the licensee should include formal presentations on the expected compliance of all stakeholders with the licensee's safety vision, including contractors, suppliers, constructors, vendors and support groups.

The operator must also establish effective relationships with the regulator, even before the bid is specified. During the early stages of an NPP deployment programme, there are many interfaces that need to be managed by the licensee, since the operator is at the centre of all the activities. The various interfaces typically include governments, regulators, the public, the media, the designer/vendor, construction companies, and manufacturers and suppliers. Notwithstanding this, the licensee and regulator must take the time to establish professional and comprehensive interactions to ensure that there is joint understanding of the licensing processes and requirements.

20.5.1 Quality assurance programme and compliance with contractual requirements

The IAEA has produced a considerable body of work on quality assurance (QA) that has been widely adopted by Member States with NPP programmes. Quality assurance in design, construction and operation is considered in detail in Chapter 21. The IAEA approach to quality programmes for NPP processes has continued to evolve to be consistent with modern approaches (Persson, 2008). Initially, quality control was established to verify the conformance of systems at the completion of a process. Then, quality assurance was implemented to focus on prevention of non-conformance during production, thus becoming more performance-based as opposed to compliance-based. Next, a quality management approach was developed to encompass everyone involved in the processes. This included the concept of corporate safety culture and a focus on people.

The most recent manifestation of the IAEA quality programmes is an integrated management system where safety, health, environmental, security, quality and economic elements of an organization are all considered together (IAEA, 2006d). This approach was designed to address two general aims stated in INSAG-13, *Management of Operational Safety in Nuclear Power Plants* (INSAG, 1999):

> To improve the safety performance of the organization through the planning, control and supervision of safety-related activities in normal, transient and emergency situations, and
> To foster and support a strong safety culture through the development and reinforcement of good safety attitudes and behaviour in individuals and teams so as to allow them to carry out their tasks safely.

Such a system is intended to produce a single coherent management system where all functions are integrated to achieve an organization's objectives, and quality requirements are incorporated fully into all the daily work. The IAEA has published Safety Guides for implementing the system (IAEA, 2006e, 2009c).

A management system for construction is also covered by these Safety Guides, particularly in Appendix V of IAEA (2009c). This Guide stipulates that an organization should develop and implement a management system that includes the overall arrangements for the management, performance and assessment of the NPP during construction and that the organization should ensure the following:

- Construction work and work at the installation are carried out in accordance with design specifications, drawings, procedures and instructions, including the implementation of the relevant requirements.
- Construction work and work that is undertaken at the installation, including work by contractors, are coordinated, carried out and completed in accordance with planned programmes.
- Access to the construction site is controlled.
- Interface arrangements exist among the construction organizations, suppliers and other organizational units performing the work.

During construction, QA includes all the actions necessary to provide confidence that a SSC will perform satisfactorily in service. This includes independent assessments of the effectiveness of all the processes related to design, procurement, and construction. The purpose of this is to ensure that the constructor delivers high-quality project work, taking into account both industrial and nuclear safety requirements. The QA plan verifies each of the processes using the hierarchy of prevention, detection, and correction. Suppliers of products and services also have to comply with the licensee's QA requirements, which could cover all the important operational areas such as procurement, materials, manufacturing, handling and storage, and shipping.

The licensee must demonstrate to the RB that the QA requirements for the construction license are being met. The RB would normally review and inspect the licensee's QA programme as well as the programmes for other involved organizations, such as suppliers of safety-related products and services, testing and calibration laboratories, nuclear steam system suppliers, and architect-engineering companies.

The licensee must also ensure that the constructor supplies all the documentation needed to define the design basis for the plant in support of operations. This involves the implementation of a comprehensive document management system that enables all records, including QA records, equipment, materials, manuals, and drawings to be controlled and maintained.

20.5.2 The pre-nuclear testing programme, results analysis and decisions

The overall purpose of commissioning is to demonstrate that the design requirements of the SSCs are met and to bring them to operating mode. Testing should establish that the NPP can operate in all the modes for which it has been designed. Commissioning is further addressed in Chapter 22.

Commissioning can be divided into four stages (IAEA, 2003d): (1) pre-operational tests; (2) fuel loading and subcritical tests; (3) initial criticality and low power tests; and (4) power tests. These stages are further subdivided into subtasks that are required by the licensee or regulator, or that depend on the technology being commissioned.

The pre-operational stage requires that construction activities associated with the system should be completed and documented, including all elements of the quality assurance programme. The construction company normally also carry out various pre-commissioning activities, such as flushing, cleaning and hydrostatically testing each system and piece of equipment individually. Also, the licensee must ensure that all equipment is ready for operation. This involves:

- Inspection of the SSCs to ensure proper construction, manufacturing and installation, such as welding, quality of workmanship, loose parts, and cleanliness
- Checking of electrical and protective devices
- Calibration of instruments
- Verification of operability of instrument loops and required response times
- Adjustment and settings of process controllers and limit switches.

Once the above has been completed, the pre-operational stage can be subdivided into two activities: cold performance tests and hot performance tests. These tests in most cases will be carried out sequentially; however, some cold tests, such as containment pressurization and leakage rate tests, might not be done until the end of the testing period, before fuel loading. Cold performance tests include the start-up of the fluid systems and support systems. The tests yield data that verify the operational functions of components and the compatibility between systems. If pressure tests on the primary and secondary systems were not previously done by the construction group, the operator will also perform these tests at this sub-stage. Hot performance tests verify that systems conform to design requirements. These tests, where practicable, should simulate anticipated operational occurrences at typical plant operating conditions.

The tests should verify the effectiveness of the various heat transport phenomena as well as checking for vibration, clearances, effectiveness of insulation, thermal expansion, and the effects of high temperature on electrical and mechanical equipment performance. Hot performance testing should be carried out at least to the point where steady-state operating conditions are reached. Completion of the initial rotation test of the turbine-generators would typically mark the end of the hot commissioning phase. Operating staff should also use this opportunity to verify the operating procedures, such as hot to cold shutdown, before fuel loading begins.

20.5.3 First fuel loading, start-up, and operation

Before fuel loading, the operators must be trained and qualified to operate the fuel handling equipment. Detailed procedures and operating instructions must be prepared and exercised during the training period with dummy fuel assemblies. Strict attention to criticality such as boron concentration levels in pressurized water reactors (PWR) is essential at this stage. Once fuel is loaded, for light water reactors (LWRs) the upper vessel internals and the pressure vessel head are installed. At this point, the operator carries out additional mechanical and electrical tests to verify that the reactivity control systems are functioning properly and reliably. The initial core monitoring system data will familiarize the operator with some practical reactor core experience.

Some additional tests are normally performed just before initial criticality to provide further assurance that the plant systems and components required for plant operation perform as expected. The plant is then brought from cold shutdown to hot shutdown to initial criticality for the start of low-power physics testing. A variety of tests are performed to confirm the core design values as used in the FSAR and other technical analyses. Reactor power is then raised through steps with test programmes at each step. The tests include physics measurements, plant shutdown and heat removal capabilities, power transients, loss of site power tests, and instrumentation and control checks.

After full power is reached and maintained for a period of time, the plant should be shut down and thoroughly inspected, and the commissioning data assessed. Any changes to the plant would be evaluated thoroughly to ensure that safety margins meet the design specifications and that the plant can perform reliably. Finally, plant acceptance testing is performed to ensure that the plant meets the contractual output. The plant operating staff typically become proficient in the operation and maintenance of components and systems during the commissioning activity.

During operation, the licensee is required to maintain detailed records concerning operations, and the nature of these records is stated in the operating license. These can include the results of effluent and environmental monitoring programmes, operating and maintenance procedures, results of the commissioning programme, results of inspection and maintenance programmes, and the nature and amount of radiation, nuclear substances and hazardous substances within the nuclear facility.

The operator must also manage plant configuration changes and the status of the SSCs over the life of the plant. A key aspect of this is the management of ageing, including both degradation and obsolescence, particularly for those SSCs important for safety. It is likely that the licensee will have to demonstrate to the regulator that it has a comprehensive and systematic management programme to address SSC ageing. The IAEA has published recommendations for the establishment, implementation, and improvement of ageing management programmes that can be used to develop an effective strategy (IAEA, 2009d). According to this guide: 'Evaluation of the cumulative effects of both physical ageing and obsolescence on the safety of nuclear power plants is a continuous process and is assessed in a periodic safety review or an equivalent systematic safety reassessment programme.' The science, technology and regulatory aspects of ageing in nuclear power plants are considered in detail in Tipping (2010).

20.5.4 Training and accreditation of operating personnel

The responsibility for safety requires that the operator establishes and maintains the necessary competencies of both staff and management for safe operations. This entails providing adequate training and effective knowledge management, establishing a culture and methodologies to maintain safety under all conditions, and verifying that all activities and processes carried out by the plant staff are safe. Since several generations of operators will likely be involved over the lifetime of the plant, which could be 60 years or longer for modern plants, knowledge management must include effective knowledge transfer mechanisms. The IAEA has published a Safety Guide that provides information on the recruitment, qualification and training of NPP staff (IAEA, 2002d). The need for human resources in establishing a nuclear power programme is considered in Chapter 6.

Operator qualifications are usually prescribed by the operating license. To approve the qualifications, the regulator requires that the licensee provide information that the licensed operator meets the applicable qualification requirements in the license, has successfully completed the relevant training programme and examinations referred to in the license, and is capable in the opinion of the licensee of performing the duties for the position. Usually, the certification is for a set period of time and must be renewed. To renew the certification, the regulator may require the licensee to provide evidence that the licensed operator has safely and competently performed the duties of the position, has continued to receive the relevant training referred to in the license, has completed the requalification tests required by the licensee, and in the view of the licensee is capable of continuing to perform the duties of the position.

A license may also require an operator to successfully complete an examination administered by the RB to become certified. The operator may take the examination only after the regulator receives from the licensee an application that includes a statement that the person has successfully completed the applicable training programme referred to in the license. The licensee is required to keep detailed records of staff training. This can include the status of each worker's qualification, requalification and training, including the results of all tests and examinations required under the license.

20.5.5 Performance improvement programme

The operator of a nuclear plant is responsible for its safety. An important operating discipline is a robust performance improvement programme. The programme should have several elements in an overall interactive model. Elements of the model could include self-assessments, operating experience feedback, conduct of operations, performance assessments, oversight standards, engineering programmes, and processes for dealing with any gaps that are identified. These elements would then fit into an overall performance model that has the following steps: (1) obtain the results for a performance monitoring/assessment element; (2) identify the gaps; (3) analyze and identify solutions; (4) implement the solutions; and (5) continue monitoring. The objectives are to identify and correct problems, to identify and correct any negative trends before they become an issue, and to raise the sensitivity of management and staff to the importance of constant diligence and questioning attitudes. While a comprehensive discussion of all the elements of a performance improvement model is beyond the scope of this chapter, two of the elements can be mentioned for illustration: self-assessment and operating experience feedback.

Self-assessment is a general process that can encompass both plant commercial performance and safety. Since performance and safety are bound together, self-assessments are a particularly important part of the overall safety structure. The IAEA Safety Requirements (IAEA, 2006d) for a nuclear facility management system mandate that: 'Senior management and management at all other levels in the organization shall carry out self-assessment to evaluate the performance of work and the improvement of the safety culture'. Individuals and workgroups must assess their performance against the licensee's safety goals including the operating license requirements and other nuclear industry safety standards. One of the values of self-assessments is that they also recognize strengths and good practices that exceed the current requirements, and these might be used to enhance performance in other areas.

A policy should be developed that lays out the objectives and procedures for performing the self-assessments. Such a policy could include the scope for the assessments, the frequency, the process roadmap, the reporting and review mechanisms, quality assurance requirements, and what benchmarks should be employed. In general, the process roadmap would involve the preparation of annual plans indicating the areas that will be assessed and the schedules, the formation of self-assessment teams to carry out the reviews, conducting and documenting the assessments, analyzing the results, taking corrective actions, and communicating the status. This process would then be followed up by an evaluation of the effectiveness and quality of the review, as well as of the lessons learned for improvement.

Two types of gaps can be identified in this way, the first being where current safety requirements, regulatory or otherwise, are not being met and corrective actions must be taken immediately, and the second being where requirements are being met but there is opportunity for improvement. In this case, although the requirements are still being met, the performance may be trending away from acceptable standards. Ideally, it is the second type of gap that would eventually come to dominate the self-assessment process. This would demonstrate that the licensee was proactively determining the precursors to any potential diminishing of safety and addressing them immediately. Therefore, self-assessments are a key activity for preventing operational complacency.

Analysis and feedback of operating experience is recognized as a valid tool to enhance safety in the IAEA *Fundamental Safety Principles*. By any measure, a nuclear power plant is a complex technology. The plant contains about 100 major systems that fall into four groups: nuclear systems, fuel and refuelling systems, secondary plant systems, and electrical systems. Due to ageing, configuration changes, and equipment upgrades, each of these systems requires verification on a continuous basis to ensure that the systems continue to meet safety and operational requirements. Operating experience, as part of the overall plant performance model, is a valuable tool for helping to ensure this, since it enables the licensee to apply previous information to anticipate and address issues before they occur. INSAG-21 has pointed out the importance of operating experience feedback for life cycle management and backfitting of nuclear facilities, as well as for improving operating and regulatory practices, to enhance the global nuclear safety regime (INSAG, 2006).

Operating experience information covers all aspects of the NPP's operation and has implications for both plant performance and safety. With respect to plant performance, particular attention is paid to outages, planned and otherwise. Outages can be classified as planned (under operator control), unplanned (causes under operator control), and external (not under operator control). Planned outages include refuelling, inspection, maintenance, testing, and upgrades. Unplanned outages include those due to human error, equipment failure, operating margins, and regulatory/licensing issues. Externally driven outages include grid failure following electricity demand, and environmental conditions.

The IAEA has established guidelines to enhance operating experience feedback (IAEA, 2006f). According to this Safety Guide, an effective system for the feedback of operational experience relating to safety should have the following elements:

- Reporting of events at plants
- Screening of events – primarily on the basis of safety significance
- Investigation of events
- In-depth analysis, including causal analysis, of safety-significant events
- Recommended actions resulting from the assessment, including approval, implementation, tracking and evaluation
- Wider consideration of trends
- Dissemination and exchange of information, including by the use of international systems
- Continuous monitoring and improvement of programmes for the feedback of safety-related operational experience
- A storage, retrieval and documentation system for information on events.

The licensee should develop a comprehensive operating experience programme with input from a variety of internal and external sources. One international tool for operating experience is the Incident Reporting System (IRS) jointly developed by the IAEA and OECD/NEA (IAEA, 2008). The IRS reports contain information on NPP events that are of significance to safety and the safety lessons that can be learned to assist in reducing recurrence of events at other plants.

More information on specific topics is also available. This includes an Information System on Occupational Exposure, which was started by the OECD/NEA and is now jointly maintained with the IAEA. Other projects at the OECD/NEA addressing specialized areas include the International Common-Cause Failure Data Exchange, the Fire Incident Records

Exchange, the Piping Failure Data Exchange, the Exchange of Operating Experience Concerning Computer Based Safety, and Stress Corrosion Cracking and Cable Ageing.

Another tool is the World Association of Nuclear Operators (WANO) series of documents on Operating Experience Report and Significant Operating Experience Report processes. However, WANO information is generally restricted to its members. There are also specific reactor technology groups, such as the CANDU, Westinghouse, General Electric and KWU Owners' Groups, which deal with design-specific operational feedback, although some information may be restricted to group members. There are also national and regional institutions which interchange operating experience.

In response to restrictions on some operating information that could impact safety, INSAG has pointed out in both INSAG-21 (INSAG, 2006) and INSAG-23 (INSAG, 2008b) that there is considerable room for improvement in the transparent sharing of safety information, both nationally and internationally. INSAG-23 also notes that:

> It is widely observed in all fields of human activity that serious accidents are nearly always preceded by less serious precursor events. If lessons can be learned from the precursors and these lessons put into practice, the probability of a serious accident occurring can be significantly reduced . . . While the continued strong safety performance by operators is encouraging, safety significant events continue to recur in nuclear installations. This indicates that operators are not learning and applying the lessons that experience can teach us.

As a result of their assessment, INSAG has proposed several recommendations to improve international operational feedback. However, implementation of the feedback still rests with the licensee. It is important to ensure that operating experience is being used effectively throughout the licensee's organization at all levels, and for both safety and operational performance. This includes the various processes for information gathering and analysis, experience application, auditing, and training.

20.5.6 Periodic safety reviews

A periodic safety review (PSR) is a comprehensive assessment of safety that is normally carried out at defined intervals as prescribed in the license. Plant ageing, configuration changes, modifications to procedures, significant events, operating experience, and other safety reviews occur over the lifetime of a plant and the PSR is a systematic way of assessing the cumulative effects of any changes to plant safety. In addition, a PSR takes into account advances in safety standards since the time of construction or the previous review. The safety of future operation of the plant can be evaluated from the PSR.

The scope of the PSR includes an assessment of plant design and operation against the current safety standards and practices. Therefore, the PSR is a tool for securing a high level of safety throughout the NPP's operating lifetime, taking into account changes in the plant and the evolution of safety knowledge. The PSR does not replace the routine safety reviews of nuclear power plant operation, which are the primary means of safety verification throughout the plant operating cycle. The IAEA provides recommendations and guidance on how to conduct the PSR (IAEA, 2003e).

From experience, the IAEA recommends that a PSR should be first undertaken about 10 years after the start of plant operation and that subsequent PSRs should be done every 10 years. The 10-year period is based on the expected developments in safety standards from both experience and ongoing R&D, and from the expected rate of the changes that could affect the plant. The PSR covers all aspects of operations, including management structures, reporting systems, staff experience and competence, plant configuration, safety culture, knowledge management, ageing effects on the SSCs, radiological protection, emergency planning, and operating experience, to mention just a few. Owing to its comprehensive scope, the PSR provides reassurance to both the licensee and the regulator that the licensing basis for the NPP is still valid.

The licensee has prime responsibility for performing the PSR. The starting point is agreement between the licensee and RB on the scope, schedule, and requirements for the review. Owing to the broad scope of the assessment, a PSR is a complex task that could take up to a maximum of three years to complete. Therefore, the IAEA has broken down the review into five subject areas with 14 safety factors to ensure that the review is comprehensive (IAEA, 2003e). These cover the plant (plant design, actual conditions of the SSCs, equipment qualification, ageing), safety analysis (deterministic safety analysis, probabilistic safety analysis, hazards analysis), performance and feedback from experience (safety performance, use of experience from other plants and research findings), management (organization and administration, procedures, the human factor, emergency planning) and environment (radiological impact on the environment). Each of these factors is reviewed and assessed against current safety standards and practices. In addition, IAEA (2003e) recommends a global assessment to integrate the results of the review of the safety factors.

Where necessary, corrective actions are determined and implementation plans are enacted. Since these actions lead to safety improvements, an objective is to complete as many of the actions as possible within the time frame of the PSR. The end point of the PSR is regulatory approval of the integrated programme to address any outstanding safety issues. Any safety gaps that cannot be reasonably addressed would require further assessment of the risk and justification to allow the plant to continue operation.

The PSR is a major undertaking that involves considerable planning and preparation. To initiate the review, the licensee establishes a dedicated project management team, develops guidance documentation laying out the scope and methodologies, defines the documents to be produced and their formats, develops a QA plan, prepares the review plan and budget, and secures approvals. The plan is then executed with many activities carried out in parallel, including, to a reasonable extent, the amelioration of issues as they are identified. This is followed by execution of an integrated plan to implement corrective actions and/or safety improvements. Further details are contained in IAEA (2003e) and recent experience in IAEA Member States is detailed in IAEA (2010b).

20.6 Regulatory compliance during design, construction, commissioning and operation

The RB must verify compliance with applicable regulatory requirements and with the license terms and conditions during each phase in the life of the NPP. Such verifications are achieved through regulatory inspections, oversight procedures, and analysis of the multiple progress and evaluation reports provided by the licensee.

20.6.1 Regulatory inspections

Regulatory inspections are carried out during each phase in the life of the NPP with the aim of checking the safety compliance of the NPP, through physical verification of the condition of its SSCs and by auditing of operational records. In many countries, the RB keeps resident inspectors in a given plant or clusters of plants to continually oversee the safety of the installation. The RB should specify the frequency, lay down formal procedures and authorize appropriate staff for carrying out these inspections. However, special inspections can be conducted when considered necessary.

The inspection findings should be discussed with senior managers of the NPP to resolve any anomalies and thereafter the inspection report should be submitted for review by the appropriate safety committees. RBs have established formal procedures on how to write inspection reports and how to formally include in such reports the licensee observations and claims to be considered before any regulatory action is taken.

The recommendations arising from the inspections and from a review of the inspection reports should be categorized according to their importance to safety following the criteria specified by the RB. Implementation of the recommendations should be done according to the schedule agreed upon between the RB and the operating organization, and the RB staff should follow this up.

20.6.2 Regulatory oversight during operation

The RB maintains a careful check on the licensee's activities during operation of the NPP to ensure that the plant is operated within the prescribed safety envelope and that other licensing conditions are complied with. This is done through review of the various operational reports, reports on safety-related incidents and on activities during refuelling outages and other extended outages at the plant. The RB also conducts periodic regulatory inspections and audits of records to physically verify the compliance of license conditions, and to check on the general upkeep of the NPP.

As discussed in Section 20.5.6 above, detailed and comprehensive PSRs of the NPP operation are undertaken by the licensee at specified intervals, typically every 10 years. The RB carefully checks the reports of such reviews to confirm that the NPP is meeting the current safety requirements and is likely to continue to meet them till the next PSR. During all these reviews, the RB should make extensive use of the operating experience from NPPs of similar design as well as other nuclear and conventional industries, as far as is applicable.

The main goal of the surveillance by the authority is to make sure that the operator follows the law and the conditions of the license. Surveillance is carried out by inspectors and assessors within the RB. There may be differences in the detailed approach undertaken in different countries, or within countries. As an example, data in paragraph A.3 of the Annex describes the activities and efforts undertaken in 2006 in Baden-Württemberg, Germany, for which an assessment of the surveillance process is available (ILK, 2006).

20.6.3 Review of operating experience and operating experience feedback

A large number of reports on the various aspects of NPP operation are generated by the licensee on a regular basis. These include reports on day-to-day operation and maintenance activities, radiological status in the plant, results of in-service inspections and surveillance checks, management of radioactive waste generated from NPP operation, chemistry parameters, and radiological surveys carried out around the NPP site. The RB should have a formal mechanism in place for an in-depth review of these reports. Initial review of the reports could be made by the RB staff and thereafter these are subjected to further review by standing safety committees constituted by the RB. Specific aspects may be referred to specialist groups for further detailed examination.

Actions identified based on the recommendations arising from these reviews should be implemented according to an agreed time schedule between the operating organization and the RB. The RB staff should follow up on the implementation of the actions meticulously. Any proposal for a change in the operating configuration of the plant must be formally submitted to the RB and should be supported by a detailed analysis which clearly establishes that the proposed change does not compromise safety in any manner. The implementation of any such change in plant configuration should be made only after a detailed review and approval by the RB.

The RB must ensure that a formal mechanism exists in the operating organization for collecting and analyzing information from the international operating experience that is relevant to the NPP for improving safety. A similar mechanism must also exist in the RB. This operating experience feedback should cover safety-related incidents at other NPPs, good safety practices adopted at other plants, and new information from research and development activities. Experience feedback from nuclear facilities other than NPPs and from conventional industries should also be considered as applicable.

While the main thrust of operating experience feedback is to prevent recurrence of safety-related events at an NPP of a nature similar to those that have occurred elsewhere, it should also be used to improve operational safety in general. In addition to actual incidents, information on 'near misses' and 'low-level events' should also be collected and analyzed for its appropriate utilization to make safety improvements in hardware and procedures.

20.6.4 Review of safety-related anomalies

All safety-related anomalies should be reported to the RB within the stipulated time frame. These should cover safety-related anomalies in operation, violation of any licensing condition or technical specifications for operation and exceeding of any prescribed limits, like those for radiation exposure of personnel or discharge of radioactive effluents to the environment. The reports should describe the incident in reasonable detail together with an analysis that identifies the apparent causes, and the root cause of the incident. They should also include the proposed corrective actions and schedule for their implementation.

If a safety limit, as prescribed in the technical specifications for operation, gets violated, the reactor must be shut down immediately and a report on the incident submitted to the RB giving details of the incident and the circumstances that caused the violation. Reactor operation can be resumed only after a detailed review of the incident and clearance from the RB. The RB should review these reports in detail according to a laid-down procedure with the primary aim of determining whether the incident occurred due to equipment failure or human error, or on account of any shortcoming in procedures or their implementation.

20.6.5 Regulatory oversight of refuelling outages and other extended outages

The NPP will have to be shut down periodically for extended periods for refuelling or for the carrying out of major maintenance work. Maintenance work is generally conducted during outages for refuelling. All activities during such extended outages should be carefully checked by the RB from a safety angle. At the end of such outages, a report should be submitted to the RB giving details of all safety-significant work done, including results of the in-service inspections and surveillance checks carried out. This report should be formally reviewed by the RB and clearance for restart of the reactor given, after confirming that the NPP meets the licensing conditions.

A large number of outside personnel, such as contractors, are likely to be engaged during such outages to carry out specific work. The RB should ensure that these personnel are given necessary training to carry out the activities, following specified radiation protection procedures and other safety requirements.

20.6.6 Regulatory review of periodic safety analysis

As explained in Section 20.5.6, the primary aim of the PSR is to assess the health of the SSCs of the NPP from an ageing viewpoint, to help ensure that there are no significant degradations that can impair safety. The RB should carry out an in-depth analysis of the PSR report submitted by the licensee to determine whether the NPP is meeting the current safety requirements and is also likely to continue to meet them till the next PSR. In addition to the information from plant data, including a revised probabilistic safety analysis, the analysis by the RB should also take into account any revision of safety standards that might have taken place, relevant new knowledge acquired from research, international operating experience and any obsolescence of NPP components. Based on this analysis a revision of licensing conditions and operating procedures should be made, as appropriate. Also the requirements for any system modification, replacement of components and other retrofitting needs should be identified and the time frame for their implementation should be decided.

20.7 Licensing of a country's first nuclear power plant

Licensing of a country's first NPP poses several challenges to the RB, as well as to the license applicant. This is mainly on account of a lack of experienced personnel who can clearly understand the safety aspects detailed in the PSAR and the FSAR and the various supporting technical documents. Other major difficulties will be the need for the licensing process to match the project schedule, and the non-availability of national safety standards.

These challenges can be met to a large extent through: (a) implementing a well-formulated human resources development plan; (b) using technical assistance from an experienced regulator (ER); (c) the use of the safety evaluation of a reference NPP that is similar in design to the NPP to be built, and which has already been licensed by a competent and experienced RB, usually one in the country of origin of the project; (d) adoption of international safety standards, mainly the ones developed by the IAEA and applied under its recommendation; and (e) development of a strategic plan for the conduct of the licensing process.

Many aspects of the national technical development to support an emerging nuclear power programme that are described in Chapter 7 are also applicable to licensing activities, for both the license applicant and the RB. Aspects specific to the licensing of a country's first NPP are covered below.

20.7.1 Human resources development

To develop its human resources, a newly established RB needs significant assistance from an ER. Accordingly, they should establish long-term collaborative links and develop a roadmap for human resources development. A core group of RB staff should receive practical training in the licensing, construction and operation at the reference NPP, as well as training from the ER in safety regulation of NPPs and use of safety standards. This core group in turn should impart training to other staff of the RB. The operating organization should similarly develop its human resources through corresponding cooperative activities with the reactor vendor.

Advanced training of selected RB staff in specific fields like reactor physics, health physics, thermal hydraulics and probabilistic safety analysis should be arranged with the ER, or with other institutions abroad. This training is generally provided by the nuclear engineering departments in universities, dedicated institutes and academies. The RB staff who will be engaged in the licensing of the NPP and its regulation during operation should study the NPP design and the operating experience of NPPs of a similar design in detail.

20.7.2 Technical assistance in design safety review

With the implementation of the human resources development actions outlined in Section 20.5.4 the RB staff can be expected to have achieved a reasonable level of technical competence to carry out the licensing review work. However, significant help from the ER will still be necessary in the licensing activities as well as in the regulation of the NPP for a few years after it goes into operation. Nevertheless, assistance from the ER during design safety reviews and in regulation of the NPP during operation should be in an advisory capacity, and the RB should assume full responsibility for all licensing decisions.

The operating organization should similarly obtain technical assistance from the reactor vendor during the various stages of licensing and also during the initial few years of NPP operation.

20.7.3 Use of safety evaluation from a reference NPP

The safety evaluation of a reference NPP, carried out at the time of its licensing, can prove very useful during the licensing of a first NPP in a new entrant country. However, it is important that the operating organization clearly understands the design of the NPP and is able to own it and defend it during the design safety review. Use of the safety evaluation of the reference NPP is not just for the purpose of speeding up the licensing process but should also lead to an enhancement of the quality of the licensing work, and the achievement of a high level of safety of the new entrant country's NPP, in an overall sense.

Some design differences between the two NPPs are, however, likely to exist on account of site specificities and plant layout. The design might also have been updated based on information from research and operating experience after the earlier licensing of the reference NPP. These differences should be clearly identified and judiciously dealt with during the design safety review. It is, however, important that the entire PSAR is subjected to a detailed review as it helps in improving the understanding of the design in the operating organization, as also in the RB.

20.7.4 Use of international safety standards

In the absence of a high level of technical competence and sufficient operating experience, it is not feasible to develop a full set of national safety standards for an NPP before the licensing process gets started. Safety standards relevant to siting of the NPP and to general safety criteria may be nationally developed from IAEA models and made appropriate to the national conditions and requirements. Other more technological standards, like those of the IAEA and of the NPP vendor's country, could be appropriately used during the detailed design safety review process. However, the safety standards required for commissioning and then for operation of the NPP should be developed before commissioning work starts. This can be done based on the knowledge of the NPP design acquired from the design safety review and with appropriate technical assistance from the ER.

The full set of national safety standards could be developed after gaining a few years of operating experience when the national experts would have also acquired a good level of technical competence.

20.7.5 Developing a strategic plan for licensing

A strategy needs to be developed for licensing of a country's first NPP to meet the project schedule, while maintaining a high level of quality in the licensing process. Significant assistance from an ER and use of the safety evaluation of the reference NPP during the design safety review are the key elements that should be appropriately included in the strategic plan.

The time available between award of contract for setting up the NPP and start of its construction are likely to be insufficient for a detailed review of the PSAR. A possible approach could be to divide the PSAR review work into suitable sub-stages. A brief review of the PSAR can be conducted, focusing on the differences in design from that of the reference NPP and ensuring that the design safety criteria are met, to ensure the award of the license to start construction.

Detailed review of the PSAR can be carried out in parallel with civil construction work at the site but should be completed before the start of activities that cannot be reversed, e.g. the erection of major equipment like the reactor pressure vessel and the steam generators. It should also be completed well before commissioning activities are undertaken. The requirements of licensing and the schedule of technical submissions by the applicant should be clearly identified in advance for each sub-stage of licensing.

20.8 Acknowledgements

The authors acknowledge John Moares, independent consultant, author of Chapter 5, 'Responsibilities of the nuclear operator in nuclear power programmes', and Erwin Lindauer, independent consultant, author of Appendix 4 'Simulator training for nuclear power plant control room personnel', for their valid inputs to the Appendix on 'Examples of licensing systems' and their comments and suggestions for this chapter.

20.9 References

HSE (2007a), *Guidance Document for Generic Design Assessment Activities*, Office for Civil Nuclear Security, OCNS 2007 Version 2, The Stationery Office, London (www.hse.gov.uk/newreactors/guidance.htm)

HSE (2007b), *Process and Information Document for Generic Assessment of Candidate Nuclear Power Plant Designs*, Environment Agency, Process and Information Document, version 1, The Stationery Office, London (www.hse.gov.uk/newreactors/guidance.htm)

HSE (2008a), *New Nuclear Power Stations. Generic Design Assessment, Guidance to Requesting Parties.* Published by the Health and Safety Executive NGN03, The Stationery Office, London (www.hse.gov.uk/newreactors/guidance.htm)

HSE (2008b), *The Management of Sensitive Nuclear Information during the Generic Design Assessment of Nuclear Technologies,* Office for Civil Nuclear Security, OCNS 2008 , version 2, The Stationery Office, London (www.hse.gov.uk/newreactors/guidance.htm)

IAEA (2002a), *Documentation for Use in Regulating Nuclear Facilities*, Safety Guide Series no. GS-G-1.4, IAEA, Vienna

IAEA (2002b), *External Human Induced Events in Site Evaluation for Nuclear Power Plants*, Safety Guide Series no. NS-G-3.1, IAEA, Vienna

IAEA (2002c), *Dispersion of Radioactive Material in Air and Water and Consideration of Population Distribution in Site Evaluation for Nuclear Power Plants*, Safety Guide Series no. NS-G-3.2, IAEA, Vienna

IAEA (2002d), *Recruitment, Qualification and Training of Personnel for Nuclear Power Plants*, Safety Guide Series no. NS-G-2.8, IAEA, Vienna

IAEA (2003a), *Site Evaluation for Nuclear Installations*, Safety Requirements Series no. NS-R-3, IAEA, Vienna

IAEA (2003b), *Seismic Design and Qualification for Nuclear Power Plants*, Safety Guide Series no. NS-G-1.6, IAEA, Vienna

IAEA (2003c), *Meteorological Events in Site Evaluation for Nuclear Power Plants*, Safety Guide Series no. NS-G-3.4, IAEA, Vienna

IAEA (2003d), *Commissioning for Nuclear Power Plants*, Safety Guide Series no. NS-G-2.9, IAEA, Vienna

IAEA (2003e), *Periodic Safety Review of Nuclear Power Plants*, Safety Standards Series no. NS-G-2.10, IAEA, Vienna

IAEA (2004a), *Flood Hazard for Nuclear Power Plants on Coastal and River Sites*, Safety Guide Series no. NS-G-3.5, IAEA, Vienna

IAEA (2004b), *Format and Content of the Safety Analysis Report for Nuclear Power Plants*, Safety Guide Series no. GS-G-4.1, IAEA, Vienna

IAEA (2005), *Geotechnical Aspects of Site Evaluation and Foundations for Nuclear Power Plants*, Safety Guide Series no. NS-G-3.6, IAEA, Vienna

IAEA (2006a), *Decommissioning of Facilities Using Radioactive Material*, Safety Requirements Series no. WS-R-5, IAEA, Vienna

IAEA (2006b), *Release of Sites from Regulatory Control on Termination of Practices*, Safety Guide Series no. WS-G-5.1, IAEA, Vienna

IAEA (2006c), *Fundamental Safety Principles*, Series no. SF-1, IAEA, Vienna

IAEA (2006d), *The Management System for Facilities and Activities*, Safety Requirements Series no. GS-R-3, IAEA, Vienna

IAEA (2006e), *Application of the Management System for Facilities and Activities*, Safety Guide Series no. GS-G-3.1, IAEA, Vienna

IAEA (2006f), *A System for the Feedback of Experience from Events in Nuclear Installations*, Safety Guide Series no. NS-G-2.11, IAEA, Vienna

IAEA (2008), *The IAEA/NEA Incident Reporting System (IRS)*, IAEA, Vienna

IAEA (2009a), *Safety Assessment for Facilities and Activities*, Safety Requirements Series no. GSR Part 4, IAEA, Vienna

IAEA (2009b), *Safety Assessment for the Decommissioning of Facilities Using Radioactive Material*, Safety Guide Series no. WS-G-5.2, IAEA, Vienna

IAEA (2009c), *The Management System for Nuclear Installations*, Safety Guide Series no. GS-G-3.5, IAEA, Vienna

IAEA (2009d), *Ageing Management for Nuclear Power Plants*, Safety Guide Series no. NS-G-2.12, IAEA, Vienna

IAEA (2010a), *Long term structure of the IAEA safety standards and current status*, December 2010, available from http://www-s.iaea.org/downloads/standards/status.pdf (accessed 3 March 2011)

IAEA (2010b) *Periodic Safety Review of Nuclear Power Plants: Experience of Member States*, IAEA-TECDOC-1643, IAEA, Vienna

ILK (2006), *Report on the Assessment of Nuclear Oversight Activities on the Minister of the Environment, Baden-Württemberg*, Internationale Länderkommission Kerntechnik, ILK-28 E, Augsburg. (www.ilk-online.org)

INSAG (1999), *Management of Operational Safety in Nuclear Power Plants*, INSAG-13, IAEA, Vienna

INSAG (2002), *Key Practical Issues in Strengthening Safety Culture*, INSAG-15, IAEA, Vienna

INSAG (2006), *Strengthening the Global Nuclear Safety Regime*, INSAG-21, IAEA, Vienna

INSAG (2008a), *Nuclear Safety Infrastructure for a National Nuclear Power Programme Supported by the IAEA Fundamental Safety Principles*, INSAG-22, IAEA, Vienna

INSAG (2008b), *Improving the International System for Operating Experience Feedback*, INSAG-23, IAEA, Vienna

INSAG (2011), *A Framework for Integrating Risk Informed Decision Making Process*, INSAG-25 (under publication), IAEA, Vienna

NEA (2004), *Nuclear Legislation: Analytical Study,* Regulatory and Institutional Framework for Nuclear Activities, 2002–2003 Updates. Nuclear Energy Agency, Paris

Persson, K (2008), *IAEA Safety Standards on Management Systems and Safety Culture*, available from http://www.efcog.org/wg/ism_pmi/docs/Safety_Culture/Apr08/IAEA%20paper%20standards%20on%20management%20systems%20and%20safety%20culture.pdf (accessed 3 March 2011)

Stoiber C, Cherf A, Tomhauser W and Vez Cardona M L (2010), *Handbook of Nuclear Law: Implementing Legislation.* IAEA, Vienna

Tipping Ph G (2010), *Understanding and Mitigating Ageing in Nuclear Power Plants: Materials and Operational Aspects of Plant Life Management (PLIM)*, Woodhead Publishing Series in Energy no. 4, Oxford, Cambridge, Philadelphia, New Delhi

20.10 Appendix: Examples of licensing systems

20.10.1 The United States

The example of the United States is relevant as it has been followed, at least partially, by many countries. Moreover, the US regulatory body has maintained cooperation agreements with many other regulatory organizations.

The Atomic Energy Act of 1954, as amended, is the centrepiece of nuclear legislation in the United States. The AEC was an independent agency charged with promoting, licensing and overseeing the peaceful uses of nuclear energy. The Energy Reorganization Act of 1974 abolished the AEC and (in 1975) created the NRC, which was given the authority to grant licenses and provide oversight of safety for nuclear civilian applications.

The NRC maintains two different approaches for licensing nuclear power plants. When the NRC was established, the decision was taken to have a two-step process linking the issuance of a construction permit, followed by an operating license. The licensing requirements under this approach are contained in the Code of Federal Regulations (CFR), 10 CFR Part 50. In 1989 the US decided to adopt a new approach (set out in 10 CFR Part 52, described further on), without abolishing the first.

Any application for a construction permit must be submitted in accordance with 10 CFR Part 50. Once received – in the form of a Preliminary Safety Analysis Report – an application is checked for completeness and formally docketed. NRC staff undertake a safety review in accordance with a Standard Review Plan (SRP) leading to a Safety Evaluation Report (SER). The SER is transmitted to a statutory Advisory Committee on Reactor Safeguards (ACRS), which provides independent advice to the NRC on the issuing of a construction permit. Before taking its final decision, the NRC has to conduct, in parallel with the safety evaluation, an environmental review of the application and prepare an environmental impact statement (EIS). At the same time, antitrust advice is sought from the US Attorney General's Office. With all this information, a public hearing is formally conducted and chaired by the Atomic Safety Licensing Board (ASLB), where interested parties may raise questions. Any dissatisfied party can request a review to the US Court of Appeals; otherwise, if the application is successful, the Director of the Office of Nuclear Reactor Regulation issues the construction permit.

The request for an operating license should be requested two to three years before the scheduled construction completion. The Final Safety Analysis Report (FSAR) is the basic document covering this phase. The main purpose of the evaluation is to check that the NPP has been built in accordance with the design approved in the construction permit and that it complies with the applicable requirements. A revision of the EIS is necessary, but neither an antitrust report nor a public hearing is conducted, unless formally requested.

The approach described in 10 CFR Part 52 was created to facilitate the standardization of nuclear power plants and simplify the two-step process by unifying the construction permit and the operation license into a single

Construction and Operation License (COL). It also introduced an early site permit and a design certification rule. The early site permit is aimed at resolving site issues, including suitability of the site for emergency preparedness and the potential existence of environmentally superior sites.

The design certification recognizes that specific designs comply with established safety regulations. Any applicant for a construction permit or operating license (under 10 CFR Part 50) or a combined license (under 10CFR Part 52) may refer to a certified design and thus ease the licensing process. As in the case with 10 CFR Part 50, the EIS, the antitrust evaluation and the public hearings are maintained.

20.10.2 The United Kingdom

The example of the United Kingdom, with one of the oldest nuclear licensing authorities, is relevant because radiation risks have not been singled out from the many other risks to which workers, the public and the environment are subjected.

The main legislation for governing the safety of nuclear installations in the UK consists of the Health and Safety at Work Act of 1974 (HSW Act), the Nuclear Installations Act 1965 (NIA65) and the Ionizing Radiation Regulations 1999 (IRR99). The organizational scheme is peculiar in the sense that radiation protection is embedded into the protection of health and safety of workers and members of the public against all types of aggressions, while in most other countries radiation is singled out as a very distinct and rather hazardous agent.

Within this context, the UK has created a chain of institutions. The Health and Safety Commission (HSC) was established by the HSW Act. Its primary function is to make arrangements to secure the health, safety and welfare of persons at work, and the public, in the way that undertakings are conducted. This includes proposing new laws and standards, conducting research, providing information and advice. The Health and Safety Executive (HSE) is the corporate body appointed to enforce health and safety law under the general direction of the HSC. The HSE is the licensing authority for nuclear installations and regulates the design, construction, operation and decommissioning of any nuclear installation for which a nuclear site license is required under the Nuclear Installations Act. Such installations include nuclear power stations. The Nuclear Safety Directorate (NSD) is a directorate within the HSE. Its mission is to secure effective control of health, safety and radioactive waste management at nuclear sites for the protection of the public and workers, and to further public confidence in the nuclear regulatory system. The Nuclear Installations Inspectorate (NII) forms the major part of the NSD. It is to the NII that the day-to-day exercise of the HSE's licensing function is delegated. The

Government has announced its intention to create a more integrated, focused, independent and accountable nuclear regulatory body. The proposal is to create an Office for Nuclear Regulation (ONR) as a stand-alone statutory corporation outside the HSE.

Any organization that proposes a nuclear installation falling within the scope of NIA65 must apply for a nuclear site license. NIA65 also states that a license can be granted only to a corporate body and that it is not transferable. It follows that the licensee must be a company, which is also a user of the site. It is important that no doubt exists about the identity of the corporate body, which has legal responsibility for the safe operation of an installation and absolute liability for injury to persons or damage to property. Where a new site is to be licensed or where an existing site is to be used for additional activities, the applicant must submit a safety case[1] to the HSE for assessment. That submission must include:

- A reference design (an initial statement of design and the safety criteria to be applied)
- A preliminary safety report (intended to show, in principle, the means by which the reference design can meet the applicant's safety criteria)
- A preconstruction safety report (a more comprehensive statement on safety analysis)
- Proposal for research and development work in support of the safety case
- Proposals for quality assurance (the means for ensuring that design, manufacture, inspection and construction are carried out reliably to the required standard)
- A contract design (the design intended for construction).

Under the UK licensing process, an operator must obtain a number of permissions before construction or operation of any nuclear installation, including nuclear power stations. The whole process starts with a generic design assessment (GDA), which allows a new power station design to be assessed before an application is made for the permissions required to build that design at a particular site. This allows early resolution of design issues arising from the assessment to be taken into account. Guidance has been provided by the HSE on how to handle the GDA (HSE, 2007a, 2007b, 2008a, 2008b).

Requests for a GDA normally originate from a reactor vendor. However, requests may also be initiated by vendor–operator partnerships. Consequently, the term 'Requesting Party' is used to identify the organization

[1] In the British terminology the expression *Safety case* should be understood as the totality of a licensee's documentation to demonstrate safety, and any subset of this documentation that is submitted to ONR for such purpose.

seeking the GDA and to distinguish it from a nuclear site license applicant. The regulators consider that it is important for potential site operators/ licensees to be engaged in the GDA process, as ultimately they will be required to demonstrate sufficient knowledge of the design before receiving permission to construct and operate a nuclear power station. The operator may also wish to be part of the design process to allow the design to be adapted to its particular needs. The generic design assessment process, referred to as 'Phase 1' in the HSE manuals, is in four stages and takes approximately 3.5 years to complete.

20.10.3 Germany

The federal structure of Germany has created a different approach which could be of interest to other countries. The Federal government has legislative power over peaceful development of nuclear power, but the licensing authority belongs to the governments of the Länder or States, which act on behalf of the Federal authority. Technical expertise is mainly held within public or semi-public entities which are called Technical Support Organizations (TSOs), as is common European practice.

Two peculiarities distinguish the German approach to licensing NPPs: there is only one time-unlimited license covering the site, design and construction, operation, substantial changes to the licensed features, and decommissioning. This license is issued in the form of partial licenses, typically about four to ten in number. This allows features which need to be constructed only in later stages to be designed in detail at a later time. In this way, the most recent technology can be used and the overall construction time may be shortened. The general design needs to be elaborated to a certain detail at the time of the first partial license. The information provided must allow the authority to make a preliminary positive statement on the whole project. It must also give reasonable assurance that the later detailed design will not result in conflicts with already licensed or even constructed features.

Another peculiarity of the German case which is of special interest is the requirement that precautions against reactor damage must be taken when deemed necessary according to the state of science and technology. This means that no fixed safety goal is given. Rather, the authority has to determine in each licensing procedure what precautions the current state-of-the-art requires. The reference to the state of science means that precautions are not limited to measures for which proven technology exists. If the state of science so requires, new technology has to be developed. The purpose of this arrangement was that, in a rapidly developing area, protection should always be in line with the most recent insights. In practice there have been regulations and standards which normally could be assumed to represent the state-of-the-art, but still the authority has had to assess whether this was indeed the case. This approach is called dynamic safety precaution.

A license must be withdrawn in case of a significant endangerment to personnel or the public, and if the remedy cannot be implemented in reasonable time. It must also be withdrawn if adequate provision for damage compensation cannot be demonstrated. In 2002, the maximum electricity production of plants was limited by law to the equivalent of about 32 operating years. The law had to be changed because any of the conditions for withdrawing a license applied. In 2010 the terms of the first agreement were changed to prolong the lifetime of the operating plants to about 40 years for older plants and about 46 years for newer ones. In both cases these changes were accompanied by an agreement between the government and plant operators. In the current situation, as a reaction to the events in Fukushima, the German government has announced the intention to accelerate the phase-out from nuclear energy by revising these lifetimes.

In the German practice, much attention is given to surveillance during operation. The Internationale Länderkommission Kerntechnik (ILK) has provided information on surveillance activities in the State of Baden-Württemberg (ILK, 2006). A so-called basic surveillance is conducted by reviewing the operator's reports and by performing inspections at the plant, and evaluating their results. This activity takes about five person-years per unit and year. The inspections are performed according to an annual inspection programme with a fixed structure but including some flexibility to take into account former performance and current problems. The programme provides inspection goals, details the items to be considered and points out the time to be spent on the various areas. In total the time spent at the plant with inspections amounts to about 48 days a year per unit. The operator is informed about the results of the inspections and the expectations of the authorities in routine meetings. In case of significant deviations, feedback is made by letter which states the actions the authority requires.

In the normal practice another part of the surveillance is reactive and generally triggered by reportable events at the plant. In the Baden-Württemberg experience a working group of individuals with different backgrounds convenes to make a first assessment, and identifies the information needed or the actions to be required from the operator. The operator's activities and reports are then followed by the department in charge of the affected unit until the authority is satisfied that the reaction taken is appropriate.

In the German practice, changes to the plant or licensed documents have to be submitted by the operator to the authority. Depending on the significance of the change, it may need an approval by the authority or a change of the license. Changes are managed by a standard procedure which includes a classification and an assessment by a TSO on the basis of which the authority decides. The work on reportable events and changes takes two to three person-years per year and unit. In performing surveillance, the authority is heavily supported by TSOs. In addition to the effort undertaken by the authority, TSOs spend some 30 person-years per year and unit. An important part of their work is the review of tests and inspections which the operator performs. This is done mainly by review of documentation and partially by attending tests and inspections. The TSOs give their assessments in the evaluation of reportable events and on proposed changes. They participate in the investigations on focal issues and review the 10-year safety reviews performed by the licensee.

21

Quality assurance during design, construction and operation of nuclear power plants

R. GASCA, Asociación Nuclear Ascó – Vandellós II, Spain

Abstract: In order to provide enough confidence that the nuclear station will produce electricity in a safe and reliable way, the implementation of quality assurance principles is necessary. This chapter identifies the main elements for establishing and implementing a quality assurance system for the stages of design, construction commissioning and operation in a nuclear power plant project.

Key words: quality, quality assurance, design, construction, commissioning, operation, management.

21.1 Introduction

The objective of this chapter is to identify the main elements for establishing and implementing a quality assurance system for the stages of design, construction, commissioning and operation in a nuclear power plant project. The content is applicable to all individuals and organizations involved in the project and the main objective of the system is to ensure and maximize safety and reliability.

In the general industrial activity, not only in nuclear, the precedent of the quality assurance process was quality control. It was based on the application of inspection and testing techniques, to verify the quality of a product against a set of acceptance criteria previously specified. The quality assurance process is based on the implementation of a set of contour conditions, affecting people, organizations and installations, to avoid or minimize deviations and to provide a reasonable assurance of getting a steady-state quality level. The quality assurance process does not eliminate quality control because critical parameters must be specifically controlled in some cases.

To better understand the role of quality assurance in nuclear safety it is convenient to introduce the concept of 'defence in depth' and its relationships.

The International Nuclear Safety Group (INSAG) of the International Atomic Energy Agency (IAEA) has established (INSAG, 1996) that defence in depth consists in a hierarchical deployment of different levels of equipment and procedures in order to maintain the effectiveness of physical barriers placed between radiological material and workers, the public or the environment, in normal operation, in anticipated operational occurrences and, for some barriers, in accidents at the plant. For the effective implementation of defence in depth the IAEA establishes that three basic prerequisites must be considered: conservatism, quality assurance and safety culture. Each level of defence can be effective only if the quality of design, materials, structures, components and systems, operation and maintenance can be relied upon. Quality assurance programmes can ensure the development of a safe design. They can also ensure that the intent of the design is achieved in the plant as built and that the plant is being operated as intended and maintained as designed.

In this chapter, the most widely applied approach for quality assurance in nuclear projects has been considered. However, the fact that nowadays a new approach called 'management system' has been established should be pointed out. This system could be defined as a set of interrelated or interacting elements that establishes policies and objectives and which enables those objectives to be achieved in a safe, efficient and effective manner.

In the area of nuclear installations this new approach has been recently introduced in the IAEA Safety Fundamentals (IAEA, 2006a) and developed in a requirements document (IAEA, 2006b). These documents define the requirements for establishing, implementing, assessing and continually improving a management system that integrates safety, health, environmental, security, quality and economic elements to ensure that safety is properly taken into account in all the activities of an organization. The system considers the implications of all actions not within separate management systems but with regard to safety as a whole.

The management system established by the IAEA includes some additional elements such as safety culture, satisfaction of interested parties and an approach to process implementation.

The IAEA has developed additional safety guides, IAEA (2006c) and IAEA (2009), to facilitate the implementation of the above-mentioned approach. Finally it should be pointed out that, for the moment, this new approach established by the IAEA is not widely applied around the world.

Coming back to the main intent of this chapter, basic criteria that are applicable to all stages of a nuclear power plant project will be identified in the following paragraphs and, afterwards, more specific elements related to the management and performance for each stage will be described.

21.2 Definitions

The following definitions of basic quality assurance terms used in this chapter have been taken from various publications (AENOR, 1995a) and two IAEA publications (IAEA, 2006b, 2007):

- *Design*. The process and the result of developing the concept, plans, calculations and specifications.
- *Construction*. The process of manufacturing, assembling, installing and erecting the structures, systems and components.
- *Commissioning*. The process by which structures, systems and components, having been constructed, are made operational and verified to be in accordance with design criteria.
- *Nuclear safety*. The achievement of proper operating conditions, prevention of accidents and mitigation of accident consequences, resulting in protection of workers, the public and the environment from undue radiation hazards.
- *Operation*. The activities performed to achieve the purpose for which the plant was constructed.
- *Regulatory body*. The authority or system of authorities designated by a State as having legal authority for conducting the regulatory process.
- *Responsible organization*. The organization having overall responsibility for the nuclear power plant.
- *Quality*. The assembly of characteristics and aspects of a product or service that make it adequate to satisfy an expectation.
- *Quality assurance*. The assembly of planned and systematic actions necessary to provide adequate confidence that an item, service or process will perform its intended function as desired.
- *Quality assurance programme*. The assembly of policies, resources and actions applied to assure the quality required.

21.3 Quality assurance criteria

In the following paragraphs the main basic criteria, applicable to all stages of a nuclear power project, will be identified and briefly described (IAEA, 2007; AENOR, 1995a).

21.3.1 Programme

A quality assurance programme shall be developed, implemented and maintained. The programme is a set of documents in which the organization establishes the overall measures to accomplish its general objectives. It will contain the organizational structure, functional responsibilities, levels of authority, role descriptions and interfaces in the activities of planning, performance and assessment.

21.3.2 Training

Personnel shall be trained and qualified in accordance with the assigned task. The training programme should have the following characteristics:

- Provide understanding of the quality assurance programme
- Describe the elements and the operation of the installation
- Provide on-the-job training
- Consider specific qualifications when required
- Ensure updating to the state-of-the-art
- Contain periodic requalification
- Require competent instructors
- Be submitted to ongoing assessment of effectiveness.

21.3.3 Deviations

All deviations from the specified criteria shall be recorded and assessed in order to identify and implement the applicable actions to solve the deviation and prevent its recurrence.

The methodology should establish measures to promptly identify, classify, analyse and correct elements, processes and behaviours that do not meet the applicable expectations. Actions to solve deviations should address the causes in order to avoid recurrences.

21.3.4 Documentation

Documents or other media which describe process or establish criteria shall be adequately prepared, reviewed, approved, issued, distributed, authorized and, as required, validated.

In the same way, records reflecting the fulfilment of quality requirements shall be specified, prepared, reviewed, approved and maintained in good condition for an established period of time. Both documents and records should be adequately stored for predefined periods of time.

21.3.5 Work management

Work shall be planned and performed in accordance with established requirements and administrative controls, and using approved documents that are periodically reviewed.

Important elements to be considered in a work management process are the following:

- Personnel competency
- Tools, equipment and materials adequacy
- Work control and supervision
- Applicable documents
- Working conditions.

21.3.6 Design

The initial design and the subsequent changes shall be carried out in accordance with established codes, standards, requirements and design bases.

The adequacy of design shall be verified and validated, before the implementation, by additional individuals or groups. Design changes should be justified and submitted to controls commensurate with the original design.

21.3.7 Procurement

Suppliers shall be evaluated and selected on the basis of specified criteria and periodically assessed. The procured items or services shall meet established requirements.

Suppliers of services acting on site should be subject to control and supervision commensurate with the safety relevance of the task performed.

21.3.8 Inspection and testing

Inspection and testing activities shall be performed under administrative controls and specified criteria. More specifically, it is necessary to establish a methodology to identify those works that require inspection or testing and the technique to be applied.

21.3.9 Assessment

The adequacy and effectiveness of the quality assurance programme shall be assessed at different scopes, levels and frequencies. More specifically, management at all levels shall regularly assess the processes for which it is responsible, in order to determine its effectiveness and identify and correct those weaknesses and barriers that hinder the achievement of quality objectives. Additionally, audits, reviews, checks and other methods of assessment, performed by personnel not involved in the work being assessed, shall be conducted on behalf of management in order to promote improvement.

21.4 Quality assurance during design

Design is the first stage of a nuclear project in which quality assurance has to be applied within the context of this chapter. The correct application, from the beginning, of the quality assurance principles will provide adequate confidence that all criteria, regulations, codes and standards have been taken into account and incorporated in the design process of safety-related systems, structures and components. This will prevent deviations, with consequences that could require difficult and expensive corrective actions, and will be the basis for safer, more reliable and efficient phases of construction, commissioning and operation.

The IAEA has established internationally accepted criteria and practices on quality assurance in design (IAEA, 1996a).

21.4.1 General considerations

The design stage of a nuclear power plant overlaps the construction stage. The responsible organization may establish separate organizations for these stages or combine them under one organization. In any case, the responsibilities and interfaces shall be clearly defined and the status of the plant established.

The design changes during all subsequent phases must be, at least, developed and implemented in accordance with the same criteria.

Additionally to the criteria identified in Section 21.3, the programme should consider aspects such as organization, interfaces, procedures, grading and human factors. In the following, some guidance on such aspects is provided.

In the area of organization and during all stages of a nuclear project, one of the more important aspects of the design control is the establishment of a single design authority. The design authority, also known as the principal designer, is the organization responsible for:

- Establishing the design requirements
- Control of interfaces
- Technical adequacy of the design process
- Ensuring that design output documents accurately reflect the design basis
- Approval of design products.

These responsibilities are applicable whether the process is conducted fully in-house, partially contracted to outside organizations, or fully contracted to outside organizations.

As for interfaces, necessary arrangements shall be established between the principal designer and the organizations involved in commissioning and operating activities. The control should be performed through workflows of information, communication channels, distribution of responsibilities and mechanisms for the resolution of problems and discrepancies.

Procedures, adequately prepared, reviewed and approved, shall define design activities such as:

- Planning
- Calculation
- Verification and validation

- Control of inputs and outputs
- Review and analysis
- Configuration control.

The application of specific quality assurance requirements may be graded considering their significance to nuclear safety. To establish the necessary grading of an item, service or process, the individual responsible should be guided through a series of questions, adapted to the case, to enable them to determine the significance, the hazards and the magnitude of the potential impact and the possible consequences in case of failure. Some examples of design activities that could be graded are the following:

- The need for and level of review and approval
- The degree of verification
- The retention time for design records
- The degree of verification and test.

Finally, the human factor shall be considered, in terms of providing a safety-conscious and stress-free work environment, so that it allows the work to be performed in safe and satisfactory conditions.

21.4.2 Specific considerations

The design process has types of activities whose specificities must be taken into account in the quality assurance system. The basic activities in the design process are the following:

- Planning
- Inputs and requirements
- Verification and validation
- Change control and outputs.

Planning

In the area of planning, every organization involved in design should plan the activities at the earliest opportunity, according to their scope, and in a chronological and documented way. The plans should include, where appropriate, the following:

- Scope of work
- Schedule of activities
- Inputs from disciplines such as safety, reliability, human factors and standardization
- Design methods
- Requirements (software, tests)

- Verification and validation activities
- Training requirements
- Controls and assessments.

Inputs and requirements

As for design inputs, procedures should be established in order to ensure that data and their modifications are adequately identified, documented, approved and controlled. Procedures should assure that data have enough detail to allow the development of the associated activities. Examples of design inputs are the following:

- Functional and performance requirements
- Applicable codes, regulations and standards
- Technical parameters such as pressure and temperature, among others
- Physical requirements such as mechanical, chemistry, electrical and structural, among others
- Requirements to prevent undue risk to the health and safety of the public
- Maintenance, reliability and test requirements
- Experience feedback
- Probabilistic safety analysis
- Human error prevention
- Interface requirements.

Analysis of design criteria should be performed in order to confirm or clarify the design basis parameters. The analysis, addressing the general criteria specified for the project, should be sufficiently detailed and documented to enable assessment by qualified personnel other than those who carried out the analysis.

Verification and validation

Design verification is a process that aims to get a reasonable assurance that the design developed fulfils all the applicable requirements, including those related to inputs, planning, design execution and control of interfaces.

Verification is performed using one or more methodologies, applied by a person or group different from that which carried out the design to be verified. Those people will have enough access to all necessary information to perform the task.

The required verification shall be performed before the affected documents are issued for purchasing, fabrication, erection or transmission to another organization in order to be used in additional design activities. When criteria cannot be reasonably fulfilled, the unverified part will be identified and controlled; in any case, the verification will be finished before the element acceptance.

The scope of the design verification depends on the safety significance of the affected element, the design complexity, the degree of normalization, the technological development status and the experience with similar previous designs.

Once a design has been submitted to a design verification process, it is not necessary to repeat it for identical designs. However, the applicability of normalized or previously approved designs, against the input data and requirements, will be verified. Additionally and if it exists, the experience feedback on normalized or previously approved designs shall be considered.

The original design and the verification activities shall be documented and traced in the records, allowing subsequent supervisions or audits on the applied methodology.

There are three methodologies to perform design verifications:

- Design review
- Alternative calculations
- Qualification tests.

In the following paragraphs the previous methodologies and validation will be described.

Design review

The design review aims to anticipate and identify potential problems or inadequacies and initiate corrective actions to ensure the final design meets the design intent. In the review process, the questions to be solved should include, but not be limited to, the following:

- Were design inputs correctly identified, selected and incorporated?
- Have original design requirements been met?
- Are assumptions adequately described and based?
- Was the design methodology appropriate?
- Were procedures followed?
- Is the design output complete and reasonable?
- Is the design output reasonable?

Alternative calculations

The verification of some kind of calculations or design analysis can be performed, comparing the original results to those obtained through other methodologies of analysis or calculation.

When alternative calculations are used to verify original calculations, reviews should be performed to confirm the adequacy of assumptions, the input data, the computer code and any other method of calculation used.

The alternative method used may be simpler or less rigorous than the original one. However, all safety-significant differences must be assessed and justified.

Qualification tests

A test programme performed on a model or prototype may be used as a design verification tool if it is performed under the most adverse design conditions for the specific design features being verified. When criteria cannot be satisfied, testing may be acceptable if the results can be extrapolated to the most adverse conditions.

Qualification testing should be performed at qualified testing facilities and in accordance with approved procedures defining the reference requirements, the test configuration and the acceptance criteria.

Design validation

This is performed after the final design verification described in the previous paragraphs, under the operating conditions of pre-operational test performed during the commissioning phase. It is carried out to confirm by examination and provision of objective evidence that an item conforms to the specified requirements.

Design outputs and change control

The final product of the design process is reflected in the design output documents that shall be adequately identified, stored and retained. A typical list of documents contains the following:

- Specifications
- Drawings
- Verification and validation records
- Technical analysis and safety evaluations.

Changes to design output, including changes to requirements, shall be justified, documented and controlled. Special consideration should be given to the impact of changes on other areas.

21.5 Quality assurance during construction

During the construction stage there are three main processes that can be developed in parallel:

- The physical implementation of the design, solving the emergent problems
- The safe, reliable and efficient development of the construction and manufacturing activities
- The installation handover for commissioning.

The large number of organizations, interfaces, activities and persons involved in this stage and under tight coordination requires a quality assurance programme to be adequately established and implemented in order to reach reasonable confidence of final success.

The IAEA has established internationally accepted criteria and practices on quality assurance in construction (IAEA, 1996b).

21.5.1 General considerations

The construction stage of a nuclear power plant overlaps other stages such as design and commissioning. The responsible organization may establish separate organizations for these stages or combine them under one organization. In any case, the responsibilities and interfaces shall be clearly defined and the status of the plant established.

The responsible organization should identify the person who will occupy the position of head of the construction organization and who will have the overall responsibility for the construction activities. That person should have enough authority and resources to assume the responsibilities of ensuring that construction and installation activities will be carried out in accordance with the applicable requirements and planned programmes.

During the construction stage of a nuclear power plant the main quality-related activities performed are the following:

- Preparing safe working procedures
- Monitoring the activities of all personnel on site
- Planning and coordinating the activities
- Controlling and supervising suppliers
- Carrying out a maintenance programme for equipment that could deteriorate
- Perform a pre-service inspection to obtain the baseline for future in-service inspections
- Arranging the handover between suppliers and organizations.

Whilst the construction organization shall retain responsibility for coordinating and planning the overall construction of the plant, suppliers should be responsible for producing detailed plans and for obtaining the approval.

Considering the number of organizations and companies usually involved in the construction phase, it is necessary that interface arrangements are agreed between participants. Examples of interfaces to be defined in writing are the following:

- Construction organization with suppliers, operating organization, principal designer, sitting organization and Regulatory Body
- Suppliers with sub-suppliers and with test and commissioning organization.

The construction stage is the previous step for the commissioning period. That is why provisions should be made by the construction organization to control and coordinate the handover of completed works between suppliers and to the commissioning organization. These provisions should include the following:

- A planned and orderly transfer of responsibilities for structures, systems and components
- That documentation of transferred items is complete, accurate and contains all non-conformances identified and solved
- Official transfer, signing of documents after a joint check of items and records.

More detailed considerations about the commissioning stage can be found in Section 21.6 and in Chapter 22.

A graded approach, based on the significance for safety, may be applied to the following activities:

- Qualification of special processes and associated personnel
- The need for, the detail and the degree of control of inspection plans
- The level of traceability.

21.5.2 Specific considerations

The construction period of a nuclear power station is characterized by two main factors: the extensive use of suppliers and the large number of items and tasks involved.

Suppliers participating in the construction must be selected from those who can demonstrate that they are suitably qualified and experienced to carry out such work. Selection will be performed by the responsible organization according to a documented process and specific criteria.

Following the award of a contract, a kickoff meeting between both parties should take place in order to review and, if necessary, clarify the requirements and identify actions. Examples of topics to be covered are the following:

- Roles and responsibilities
- Interfaces and methods of communication

- Documents to be approved before use
- Training and qualification
- Materials and equipment
- Processes and expectations
- Records
- Assessments.

The use of sub-suppliers should be approved by the responsible organization case by case and after the identification of the necessary arrangements to ensure that the requirements are fulfilled.

During the construction stage, a large number of items are received, stored, handled and used. To prevent their damage, misuse or loss of traceability, items should be controlled. Items arriving on site should be visually inspected before unloading to verify the absence of damage. After receipt, a more detailed inspection will be performed to assure compliance with the applicable requirements. Examples of elements to be inspected are:

- Manufacturing and test documentation
- Identification (traceability)
- Configuration
- Protection
- Damage
- Cleanliness.

After reception, items will be stored under conditions specified to prevent any damage prior to their installation or use. Storage areas should be established and controlled, considering aspects such as:

- Access and security
- Safety grades
- Cleanliness and housekeeping
- Identification
- Protection
- Preventive maintenance
- Limited service or shelf-life
- Physical and chemical characteristics.

At any time, all items must be handled by competent personnel, using adequate equipment and taking into account aspects such as:

- Weight and size
- Prescribed handling points
- Susceptibility to shock damage
- Handling equipment status and requirements
- Maintenance of environmental conditions
- Preservation of protection.

To adequately control the activities of construction, the responsible organization will establish a supervisory programme containing the following elements:

- Methods
- Schedules
- Level required
- Acceptance criteria.

21.6 Quality assurance during commissioning

The main objective of the commissioning period is to demonstrate that the nuclear power plant has been constructed and functions according to the design intent and therefore the operation stage may start. An adequately established and implemented quality assurance programme will provide confidence of the fulfilment of such goal.

Commissioning is specifically considered in Chapter 22 so that the items covered in the previous and the following paragraphs are considered in a much broader sense, not only from the quality assurance point of view.

The IAEA has established internationally accepted criteria and practices on quality assurance in commissioning (IAEA, 1996c).

21.6.1 General considerations

The main objective of the commissioning period is to demonstrate that the nuclear power plant has been constructed and functions according to the design intent and, in consequence, that the operation stage may start.

The commissioning stage of a nuclear power plant overlaps two other stages: design and operation. As in previous stages, the responsible organization may establish separate organizations for these stages or combine them under one organization. In any case, the responsibilities and interfaces shall be clearly defined and the status of the plant established.

The responsible organization should identify the person who will occupy the position of head of the commissioning organization and who will have the overall responsibility for the commissioning activities. That person should have enough authority and resources to assume the responsibilities of executing the commissioning programme and operate systems and components as necessary.

In this stage, the responsible organization shall establish for the first time a set of important programmes that will have continuity during the operation stage:

- The radiation protection programme to protect the workers, the public and the environment against undue risks

- The security programme to control access, prevent intrusion and avoid damage
- Emergency planning and preparedness to manage emergency situations.

The current industrial safety programme will be modified, according to the results of the new risk analysis, adapted to the new plant status.

There are many internal and external interfaces that the responsible organization should address by applying procedures. Examples of interfaced organizations are the following:

- Construction organization
- Principal designer
- Operating organization
- Regulatory body
- Suppliers
- Inspection agencies.

Commissioning activities, whose requirements could be graded according to their safety significance, are the following:

- Component testing
- Test analysis
- Commissioning records
- Equipment calibration programme.

The commissioning stage is the previous step for the operating period. That is why provisions should be made by the commissioning organization to control and coordinate the transfer of the whole plant on completion of commissioning activities. Before the commissioning activities are considered completed, all deviations shall be resolved.

During this stage, not only the specific activities related to commissioning shall be planned, but those related to modifications, replacements, preventive maintenance and repair.

Programmes or methodologies established in previous stages will be adapted to consider the commissioning activities and requirements, as follows:

- Documentation and records
- Procurement
- Handling and storing
- Measuring and test equipment
- Housekeeping
- Training.

21.6.2 Specific considerations

During the commissioning period, specific consideration should be given to the following areas that will be described below:

- Programme
- Design control
- Commissioning procedures
- Component and system control
- Verification of commissioning activities.

Programme

To establish the scope of the commissioning programme, the plant should be divided into components, structures and functional systems. For every item or group of items, a detailed test programme will be prepared considering the design, function and performance characteristics.

During the testing programme, the following important data, which will be useful in the operating period, should be collected and documented:

- Operating parameters
- As-built characteristics
- Operating set points.

Design control

Any design change required as a result of commissioning activities shall be submitted to the design authority in order to be reviewed and approved, applying the same principles and methodology that were applied to the original design. The test results shall also be reviewed by the principal designer to ensure that equipment functionality is acceptable.

Commissioning procedures

The commissioning procedures should contain the necessary information related to applicable requirements, test objectives, personnel and equipment requirements, precautions, step-by-step instructions, acceptance criteria and records to be generated. Additionally, the principal designer should be involved in the procedure review process.

Component and system control

To ensure that systems and components are adequately identified and that their related documentation is traceable, a methodology should be developed and implemented. It is desirable that this identification methodology could be transferred to the operations stage without changes.

In the commissioning period and in order to avoid events that can produce personnel injuries or equipment damage, it is necessary to control the operational status of structures, systems and components. An adequate methodology shall be deployed to establish whether an element is in service, in testing or out of service and to change an element's status in a controlled manner.

When components and systems are taken out of service or returned to operation, verification should be provided to the extent necessary to ensure that elements are in the desired status.

In some cases, for example to perform a test or to solve a problem, it could be necessary to implement a controlled temporary modification to the installation. These modifications shall be risk-analysed, before the implementation, in order to determine their viability and the need for compensatory measures or contingency plans. Status control shall be applied to the elements affected by this process.

Verification of commissioning activities

Some commissioning activities shall be verified, applying methods and acceptance criteria described in approved procedures. Examples of such activities are the following:

- Accordance between test pre-requirements and test procedures
- Parameters within the proper ranges for test conditions
- Reviews conducted as required
- Hold points fulfilment.

The principal designer should be involved to verify that testing is in accordance with design intent (procedure) and that design requirements have been met (results). When a design requirement is not fulfilled, a non-conformance notice should be issued to manage the situation.

21.7 Quality assurance during operation

If decommissioning is not considered, operation is the last and longest stage of a nuclear project. The technical aspects of decommissioning are considered in Chapter 24.

In this period, the quality assurance programme must retain criteria applied in previous stages, because design, construction and commissioning are still present to a lesser extent but with the same relevance. Additionally, the programme must have an operational focus to consider the three basic facts that characterize the nuclear generation of electricity:

- The large amount of energy stored in the reactor
- The necessity of removing the reactor's residual heat for a long period of time
- The manipulation of radioactive products.

Safe and reliable operation of a nuclear power plant cannot be achieved without a sound quality assurance programme adequately established and implemented.

The IAEA has established internationally accepted criteria and practices on quality assurance in operation (IAEA, 1996d).

21.7.1 General considerations

The operation stage starts by overlapping with the commissioning period and ends with the decommissioning activities.

Nuclear power plant structures, systems and components must be formally transferred from the commissioning organization to the operating one, assuring the following aspects:

- Components are checked to verify aspects such as identification, integrity, completion of tests and inspections, alignment, calibration and housekeeping
- Deviations are resolved
- The documentation is correct and complete and reflects the as-built condition.

The operating organization should identify a person, endowed of the necessary authority and resources, to be responsible for ensuring that all activities are performed so as to assure the safety of the public, personnel, plant and equipment, in general, and more specifically to assure that all activities are carried out in accordance with the regulatory requirements.

In the operating period there are many organizational interfaces that should be formally addressed in documents. Examples of such interfaces are:

- Commissioning organization
- Regulatory body
- Off-site organizations
- Organization responsible for design
- Plant departments
- Operating shifts.

As in other stages previously described, a graded approach to the quality assurance requirements, based on the relative importance for nuclear safety, may be established in activities such as:

- Level of detail in the operating and maintenance instructions
- Testing, surveillance and inspection
- Reporting level of deficiencies
- Calibration and condition monitoring
- Documentation and recording.

In this stage, the operating organization should adapt the programmes developed in the commissioning period (for instance radiation protection, waste management and emergency preparedness) to the new circumstances. Additionally, the operating organization shall establish a fire prevention and protection programme to protect personnel and equipment, providing methods and means for preventing, detecting, controlling and extinguishing fires, and requiring periodic drills and exercises to confirm the programme's degree of implementation and effectiveness. This programme, which should be consistent with the industrial regulations, should also contain adequate measures for controlling generation, and storage of combustible materials.

In the area of human factors, the working environments should allow work to be carried out in a safe and satisfactory way. Those factors that could influence the effectiveness and the fitness for duty of the personnel should be identified and addressed. Examples of such factors are the following:

- Duration of work time
- Availability of resources to perform and supervise the works
- Local conditions such as lighting, humidity or temperature
- Adequacy of alarms in terms of number, position and prioritization
- Availability of adequate procedures, tools and equipment.

21.7.2 Specific considerations

During the operation stage there are many specific aspects to be considered. The most important are grouped here in four basic areas: organizational process, documents, installation and personnel.

Organizational process

Under the basic area of organizational process, elements such as planning, verification, testing, fuel handling, waste management, maintenance, chemistry, in-service inspection and operating experience will be considered.

A planning system, adequately established and implemented, is necessary to ensure that work at a nuclear power plant is planned and completed in a safe and reliable manner. The system should identify elements such as the following:

- The work to operate and maintain the station
- The relative importance of each task through a graded approach
- The instructions to perform the work
- The requirements related to aspects such as radiation protection, fire prevention and testing
- The records required to document the task performed
- The personnel requirements to perform the work.

A very important period in the operating stage is the outage of the plant for refuelling, maintenance or modification. In these cases, detailed planning and tracking systems are required to ensure controlled execution of activities. Outage planning is a continuing process involving the next scheduled outage and several future outages. The outage plan should include an overall plan to control and properly sequence outage tasks and provide sufficient detail to coordinate the work and track the progress.

Fuel handling is performed, depending on the basic design, while the plant is operating at power or in outage. In any case, that activity must be carried out under controlled conditions from the time of receipt of fuel through core loading, approach to criticality, on-line refuelling, and fuel removal, storage, transportation and disposal.

In order to identify and correct human errors that could produce events, verification activities could be required in some operating items, services and processes such as restoration after maintenance or testing, relevant operating manoeuvres or availability of standby elements.

To assure availability and reliability of structures, systems and components, tests are conducted during the operational phase. These tests can be divided in two groups: surveillance tests and functional tests. Surveillance tests are performed periodically at a pre-established frequency, and functional tests are performed after maintenance, repair or modification. In any case, tests shall be performed following sufficiently detailed procedures describing instructions, acceptance criteria and records.

During power operation radioactive waste generation cannot be avoided but should be minimized and provisions made for the safe handling, storage, transport and disposal of radioactive solids, liquids and gases. The control activities should ensure that radioactive wastes are within the authorized limits and conditions established by regulations in aspects such as identification, segregation, activity level, packing and records.

During the pre-operational stages, the operating organization shall prepare a maintenance programme based on pertinent information from designers, manufacturers and other operating organizations. The basic elements of a successful maintenance programme are planning, qualified personnel, procedures, spare parts, special tools and equipment, and working environment.

In general, there are two kinds of maintenance: corrective maintenance, understood as the repair and restoration of defective items, and preventive maintenance that tries to avoid or predict the failure. In both cases, the results must be recorded, trended and assessed in order to identify and implement improvements that increase availability and reliability.

Another important process is to provide optimum protection for plant systems through an adequate control of chemistry and radiochemistry that minimizes the corrosion process and the build-up of radioactive products. The main activities in the chemical and radiochemical process are the following:

- Timely detection and correction of abnormalities through sampling, monitoring and trending parameters
- Identifying and correcting deficiencies and errors through data evaluation
- Ensuring the accuracy of analytical methods through the control of analytical conditions
- Ensuring the proper management of chemicals.

One of the best ways to improve safety and reliability is to learn from one's own mistakes and, even better, from others' mistakes. That is why the operating organization should implement an internal and external operating experience programme considering four basic elements:

- Capture of internal and external information
- Screening to identify the applicable information
- Analysis of selected events in a graded approach
- Identification and implementation of corrective actions.

Documents

Before the beginning of work, adequate documents should be available in order to allow the people to perform the task in a safe and reliable way. Documents provided by vendors and drawings containing acceptance criteria could be acceptable if the applicable sections are identified in the plant documents.

To determine the degree of detail in a working document, the personnel competencies and the specific characteristics of the work should be considered. Additionally, the working documents should provide enough flexibility to accommodate variations in the work methods while identifying the applicable limitations in technical or managerial areas.

In the area of working documents, special consideration should be paid to procedures and more specifically those related to normal operation, emergency operation and temporary activities.

In the operation stage, operating procedures should be provided for the following activities:

- Transition from refuelling outages, through the different reactor conditions, until power operation
- Steady-state power operation
- Changing load
- Response to abnormal conditions and alarms
- Fuel loading and unloading
- Operational testing.

Procedures for dealing with emergency conditions should be prepared to be used with the objective of returning the plant to conditions covered by the normal operating procedures or at least to provide a safe shutdown state for a long period of time. Due to the unexpected characteristics of an emergency situation, procedures should provide enough flexibility to cope with changes in the situation, including multiple and sequential failures.

Sometimes a permanent procedure to perform an unusual or a brief task is not available. In these cases the use of temporary procedures is acceptable, providing the following conditions are met:

- The same control requirements as for permanent procedures
- Identification of the period of time during which they may be used
- Periodic assessment to confirm their need.

Installation

Under the headline of installation will be considered aspects such as housekeeping, element identification, equipment status, and temporary modifications.

Maintaining plant housekeeping and cleanliness is an essential activity to prevent negative circumstances such as:

- Entrance of foreign materials in open systems
- Contamination of items
- Uncontrolled movements of elements and personnel in and out of work areas
- Accidents and injuries.

Additionally, plant areas, structures, components and systems shall be uniquely and permanently labelled to facilitate positive identification to personnel. This identification, consistent with the codes and terminology used in the project documents, is the basis for adequate equipment status and control which will reduce the probability of mistakes that can lead to events.

The control room operators shall be kept permanently informed of the plant status. Structures, systems and components can be in different situations such as in operation, available to operate, out of service for some reason, under testing, or affected by an abnormal or limiting condition. To achieve that objective, a methodology of configuration control must be implemented. This methodology will take into account the following considerations:

- Control measures, such as locking and tagging, should be documented and used to prevent injuries and accidents.
- The position of valves, switches and other important items shall be known.
- Procedures describing the work authorization process should clearly define responsibilities on equipment isolation, post-maintenance testing and return to operation.
- Placement and removal of tags shall be controlled.
- Operating personnel will grant permission for work after assessing the situation in particular and the plant status in general, during the forecasted period of time.
- Depending on the circumstances, the removal of an element and its return to service should be verified.
- When elements are returning to service, operating personnel should confirm its functional acceptability.

During the operation stage, and due to different circumstances, it could be necessary to implement temporary modifications such as electrical jumpers, bypass lines, temporary settings, lifted electrical leads, temporary blank flanges and temporary defeats of interlocks. The documented methodology to control temporary modifications shall take into account the following considerations:

- Assessing and approving the modification before implementation
- Periodic review to confirm their need
- Minimization in number and time limitation.

Personnel

In general, personnel are expected to apply continuously a set of safe attitudes and behaviours, such as awareness, attention to detail, a questioning attitude and conservative decision making.

As a basic principle, operating personnel are responsible for operating the plant in accordance with operational limits and conditions. To fulfil this expectation, it is necessary that operators are informed of all activities performed on the installation that could affect safe and reliable operation.

More specifically, the people working in the control room should apply the following practices:

- Acknowledging, analysing, responding to and eliminating the causes of alarms
- Maintaining the plant logs
- Acting professionally in the control room
- Optimizing the amount of paperwork to be performed during the shift
- Keeping other personnel informed about operating activities in progress
- Being prepared for emergency situations.

Line managers and supervisors, as part of their daily activities, should review the conduct of work under their responsibility. The main activities related to this responsibility are the following:

- Keeping informed about plant conditions
- Monitoring work
- Ensuring that deviations are identified and solved
- Being alert to improvement opportunities
- Evaluating plant operation and documents
- Assisting the planning of future work.

In the field, and additionally to those mentioned previously, supervisors should foster the implementation of practices that promote a safe and reliable operation. For example:

- Use of error reduction tools such as self-checking, peer checking, three-way communication, use of phonetic alphabet, pre-job briefing and use of procedures
- Industrial safety, security and radiological protection practices.

During the operation stage, the responsible organization works 24 hours a day, seven days a week; in other words, people work on shift. In this kind of organization a relevant process is shift changeover and, like any other process, it must be formalized in terms of defining elements such as persons involved, distribution of responsibilities, location, methods of information transfer and provisions for special circumstances. More specifically, shift turnover should address the following issues:

- Operating status of structures, systems and components
- Relevant parameters
- Abnormal or degraded conditions
- Significant works in progress
- Work planning
- Special or temporary instructions

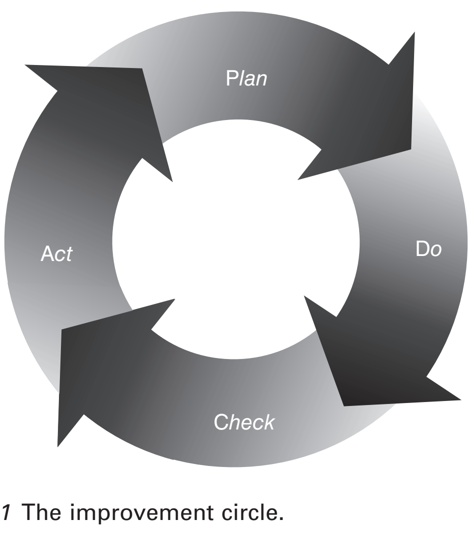
- Log readings
- Alarm status
- Key safety parameter status
- Relevant trends.

The participation of external personnel in the operational stage is less frequent than in the previous stages but could be more relevant. In fact, personnel who are not dedicated to the specific nuclear power plant areas and personnel of contracted suppliers who perform activities on plant systems should be appropriately trained and qualified for the work they are to perform. Considering their previous training and qualification, enough time should be provided to receive general employee training and specific training on the applicable plant procedures and practices. The work of these personnel should be reviewed by plant supervisors.

21.8 Assessment

Taking as a reference the classic quality improvement circle 'Plan – Do – Check – Act', represented in Fig. 21.1, the first two steps, 'plan' and 'do', have been covered in the previous paragraphs since criteria to define, establish and implement a quality assurance programme have been identified. The objective of this paragraph, devoted to assessment, is to cover the steps related to 'check' and 'act'.

The IAEA has established internationally accepted criteria and practices on the assessment of the implementation of a quality assurance programme (IAEA, 1996e).



21.1 The improvement circle.

21.8.1 General considerations

Assessments could be defined as those activities performed to verify the implementation of applicable requirements and the adequacy and effectiveness of the process. The result of an assessment is the identification of areas for improvement and the implementation of actions to solve or prevent a deficiency.

As with many other activities covered by the quality assurance programme, assessments shall be performed by personnel adequately qualified and the scope can be graded based on the safety relevance of the issue.

In general, an assessment should be performed following five basic steps and activities:

- Planning: selection of areas, activities and requirements to be assessed
- Conduct: observation of activities, interviews and review of records
- Evaluation: identification of findings considering the causes
- Reporting: written communication of the performed activities and findings
- Follow-up: identification and implementation of corrective actions by the assessed organization. Verification of implementation of the actions and close of the assessment by the assessor.

The different kinds of assessments can be classified according to the following schema:

- Independent assessments:
  - External assessments
  - Internal independent assessments
- Self-assessments:
  - Management self-assessments
  - Supervisors and workers self-assessments.

21.8.2 Specific considerations

In the following paragraphs some guidance on self-assessments and independent assessments will be provided.

Independent assessment

This is a kind of assessment that is performed by personnel without responsibility on the assessed matter. They can be divided into external or internal depending on the precedence of the assessors. Examples of external assessments are:

- Peer reviews performed by the World Association of Nuclear Operators (WANO)

- Reviews performed by the International Atomic Energy Agency (IAEA) through an Operational Safety Assessment Review Team (OSART)
- Reviews performed by qualified agencies to certify the implementation of international standards such as ISO 9001 (quality management systems) or ISO 14001 (environmental management systems).

The internal independent assessments are performed by specific assessment units that have been designated by the responsible organization with the objective of acting on behalf of management. Examples of typical subjects addressed in internal independent assessments during different stages are as follows:

- Design:
  - Use of software
  - Design reviews
  - Calculation control
  - Document control
  - Use of models
- Construction:
  - Suppliers
  - Housekeeping
  - Control of handover process
  - Materials testing
  - Control of non-conformances
- Commissioning:
  - Organizational interfaces
  - Safety management
  - Housekeeping
  - Labelling
  - Equipment status control
  - Testing programme
- Operation:
  - Control room activities
  - Operating experience feedback
  - Equipment reliability
  - Design and procedure changes
  - Control of abnormal conditions
  - Radiological protection
  - Security.

In the operational phase, a widely applied practice is to assess all safety-related areas every two years.

Self-assessment

This is a kind of assessment performed, in a routine and continuing process, by personnel with some degree of responsibility on the assessed matter. That includes technicians, supervisors and managers.

The activities of self-assessment are very varied. Good practices of self-assessment performed by shopfloor workers are:

- Self-checking, performed in risky activities or situations, through the application of four steps: stop, think, act and review
- Peer observations, performed during work stoppages to assess the fulfilment of expectations such as industrial safety requirements.

For supervisors, self-assessment activities are basically discrete checks such as inspecting, testing and checking.

In the case of line and senior managers examples of self-assessment areas and tools are:

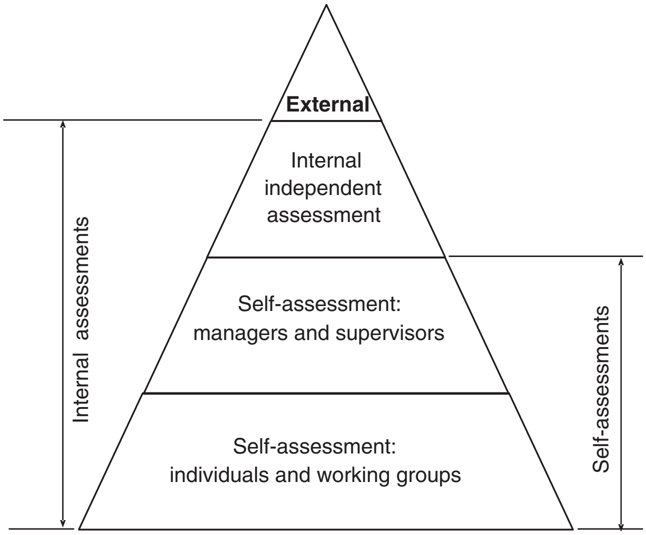
- Objectives accomplishment
- Results and trends
- Observation of processes and services
- Plant tours
- Operating experience feedback.

To improve a process it is necessary to have reliable information about it or, in other words, to measure its basic parameters. The operating organization should identify and monitor the parameters that provide information about safety, reliability and effectiveness. These parameters, named performance indicators, are useful as an assessment tool in the following ways:

- Establishing acceptance criteria (thresholds) that require corrective actions if overridden
- Identifying trends and recurrences
- Comparing performance to that of other organizations.

Good examples of internationally accepted performance indicators are those developed by the World Association of Nuclear Operators covering areas such as:

- Production data
- Reactor trips
- Availability of front-line safety systems
- Fuel reliability
- Collective dose
- Chemistry parameters
- Industrial safety rates.



21.2 The assessment programme.

A graphical representation of the assessment programme's structure would have a pyramidal shape, as in Fig. 21.2, where self-assessment would be on the bottom, because of its comprehensiveness and the amount of resources involved. For the same reasons, the internal independent assessment would be in the middle of the pyramid and the external independent assessment would be on the top.

A good way to improve the assessment programme is to answer the following question, each time an assessment tool detects a deficiency: why didn't the lower assessment level detect it?

21.9 Human resources

Human resources requirements and training programmes are the subject of Chapter 6. In this section, some specific orientations on the quality assurance area, applicable to any stage of a nuclear project, are provided.

In a nuclear project, everyone is responsible for achieving and maintaining quality, through the implementation of a quality assurance programme. This is why the existence of a specific unit devoted to quality affairs is not specifically required. However, an organization devoted to internal independent assessment activities should exist.

These units usually receive names such as quality assurance, quality management, assessment and others. Their size may differ widely from one project to another depending on factors such as regulations, general policy and organizational structure. However, they should have three basic common characteristics:

- Free access to any activity, organization, individual or document affected by the quality assurance programme
- Enough independence from other project organizations
- Strong support from top management.

Once these previous general criteria are established, the next two subsections will provide specific guidance on training for people performing internal assessments or involved in the quality assurance programme implementation.

21.9.1 Personnel performing quality-affecting activities

AENOR divides these personnel into three groups: managers, technicians and workers (AENOR, 1999).

Managers having the final responsibility for establishing and implementing the quality assurance programme should be thoroughly instructed and trained on the fundamentals and requirements of the programme, the historical development of management and quality assurance systems and the codes and basic applicable regulations. Human factors such as communication, motivation and leadership should be considered as well.

Technicians having responsibility for execution, leadership or supervision of quality-related activities should be knowledgeable of the applicable fundamentals and requirements of the programme. In that way they should receive training and instruction on the following aspects:

- Codes and regulations
- Criteria related to management aspects of the establishment and implementation of the programme
- Corrective action programme
- Quality assurance requirements applicable to the stage of the project, for example siting, design, procurement and fabrication, construction and erection, licensing and operation
- Techniques on assessment or verification such as inspection, testing, statistical methods, supervision or audits
- Human factors.

Workers having responsibility on the direct execution of quality-related activities according to established procedures will be instructed and trained on the following aspects:

- Applicable requirements of the quality assurance programme
- Methodologies to identify and report deficiencies
- Use of error-prevention tools.

21.9.2 Personnel performing assessment activities

Personnel performing assessments should be trained in quality assurance principles and in the specific methodology. Criteria for qualification of assessment personnel should be established and include technical knowledge, professional competence and experience. The assessment personnel should also have the ability to effectively observe, evaluate and report. Communications skills, integrity and the ability to maintain confidentiality and objectivity are desirable attributes (IAEA, 1996e).

The assessment personnel should maintain their proficiency and technical knowledge, applying actions such as:

- Regular participation in assessments
- Study of related documents
- Participation in training courses and seminars.

Different levels of qualification could be established for personnel performing assessments. The levels should be associated with the roles in the assessment (execution, supervision or preparation) and the requirements, in terms of academic certifications, experience or training, and should be graded accordingly. In the case of supervisions or reviews three levels could be acceptable. In the particular area of audits two levels are generally accepted: auditor and lead auditor.

21.10 Sources of further information and advice

There are international organizations and national institutions that have long been very active in developing quality assurance principles and requirements. The work and publications of such organizations and institutions provide detailed information on the subjects presented in this chapter.

Among others, three international organizations are considered good sources of further information. As already seen in the text, the IAEA is specifically concerned with quality assurance at nuclear installations. The International Organization for Standardization, ISO, is a large organization dealing with all types of standards, including the subject of quality assurance from a holistic viewpoint. Within Europe, the European Foundation for Quality Management, EFQM, emphasizes the basic aspects of managing quality assurance.

All industrial nations have national institutions for quality assurance and quality management in general, as well as working groups or specific institutions to consider nuclear activities. It is not possible to record all these national activities. Three countries have been selected: the USA, Germany and Spain. The first two are heavily industrialized and have developed whole sets of nuclear standards on quality assurance directly applicable to nuclear power plants; although the basic principles are similar, the approaches are different in accordance with the idiosyncrasy of the corresponding industrial practices. Spain has been an importer of nuclear technology from both countries, has assimilated both approaches and created its own set of quality assurance requirements. It could be a good example for those entrant countries that may rely on different exporters for their nuclear power projects.

21.10.1 International institutions

International Atomic Energy Agency (IAEA)

The Agency is an independent international organization and part of the United Nations family. Three main pillars – or areas of work – underpin the IAEA's mission: Safety and Security; Science and Technology; and Safeguards and Verification. The IAEA documents selected for further information are as follows:

1. *Establishing and Implementing a Quality Assurance Programme*, 50-SG-Q1, 1996.
2. *Non Conformance Control and Corrective Actions*, 50-SG-Q2, 1996.
3. *Document Control and Records*, 50-SG-Q3, 1996.
4. *Inspection and Testing for Acceptance*. 50-SG-Q4, 1996.
5. *Quality Assurance in the Procurement of Items and Services*, 50-SG-Q6, 1996.
6. *Quality Assurance in Manufacturing*, 50-SG-Q7, 1996.
7. *Responsibilities and Capabilities of a Nuclear Programme Implementing Organization*, NG-T-3.6, 2009.
8. *The Operating Organization for Nuclear Power Plants*, NS-G-2.4, 2001.
9. *OSART Guidelines*, 2005.
10. *Application of the Management Systems for Facilities and Activities*, GS-G-3.1, 2006.
11. *The Management System for Nuclear Installations*, GS-G-3.5, 2009.
12. *Nuclear Power Plant Outage Optimisation Strategy*, TECDOC 1315, 2002.
13. *Management Strategies for Nuclear Power Plant Outages*, TRS 449, 2006.

International Organization for Standardization (ISO)

ISO is a non-governmental organization that develops and publishes international standards. Two ISO documents, ISO (1996) and ISO (2000), selected for further information, describe criteria and requirements for the implementation of systems to manage the areas of quality and the environment.

European Foundation for Quality Management (EFQM)

The EFQM is a non-profit membership foundation. Its mission is to help organizations to continuously improve and achieve higher levels of performance, providing different kinds of services. The document selected for further information, EFQM (2010), defines an *excellence model*. The model establishes eight fundamental concepts of excellence and a framework based on nine factors, five of them being 'enablers' and the rest 'results'.

21.10.2 United States of America

Nuclear Regulatory Commission (NRC)

In the USA, all persons and organizations who receive a licence from the Regulatory Body, NRC, to use nuclear materials or operate nuclear facilities are obliged to comply with Title 10 of the Code of Federal Regulations (CFR). This Code is divided in parts and part number 50 is devoted to '*domestic licensing of production and utilization facilities*'.

Appendix B of Title 10, part 50, of the Code of Federal Regulations (10 CFR Part 50) establishes 18 basic quality assurance criteria for nuclear power plants and fuel reprocessing plants.

The NRC has also developed criteria to guide their staff in the review of applications to construct and operate nuclear power plants. All these criteria are compiled in a document identified as *Standard Review Plan*, NUREG 0800. It contains 18 chapters, one of them fully devoted to the applicable requirements in all stages of a nuclear project.

American National Standards Institute (ANSI)

The US standards and assessment system is based on the work of the American National Standards Institute (ANSI). The ANSI assigns overall responsibility for coordination, development and maintenance of nuclear power quality assurance standards to the American Society of Mechanical Engineers (ASME). The ASME Committee on Quality Assurance has prepared the following document (ANSI/ASME, 2008) as a source of further information:

- ANSI/ASME NQA-1, *Quality Assurance Requirements for Nuclear Facilities Applications*, 2008.

In this document, specific requirements for the 18 criteria identified in Appendix B of 10 CFR Part 50 are developed.

21.10.3 Germany

The Nuclear Safety Standard Commission (KTA) is the organization responsible for issuing standards on nuclear technology in Germany. Among a programme of nearly 100 standards, general requirements (KTA, 1996) establishes general criteria applicable to 10 areas in all stages of a nuclear project. Other KTA standards have been selected as sources for further information and are as follows:

1. *Requirements Regarding the Operating Manual*, KTA 1201, 1985.
2. *Requirements Regarding the Testing Manual*, KTA 1202, 1984.
3. *Documentation during the Construction and Operation of Nuclear Power Plants*, KTA 1404, 1989.

21.10.4 Spain

Consejo de Seguridad Nuclear (CSN)

The Spanish Nuclear Regulatory Body, CSN, has established a set of safety guides to describe acceptable methods to fulfil the Spanish nuclear regulations. There is a safety guide (CSN, 1999) of a general nature and a series of specific guides (in Spanish only) as follows:

1. CSN, *Sistema de Documentación sometida a Garantía de Calidad en Instalaciones Nucleares*, GS 10.2, 2002.
2. CSN, *Auditorías de Garantía de Calidad*, GS 10.3, 2002.
3. CSN, *Garantía de Calidad para la Puesta en Servicio de Instalaciones Nucleares*, GS 10.4, 1987.
4. CSN, *Garantía de Calidad en Procesos, Pruebas e Inspecciones de Instalaciones Nucleares*, GS 10.5, 1999.
5. CSN, *Garantía de Calidad en el Diseño de Instalaciones Nucleares*, GS 10.6, 2002.
6. CSN, *Garantía de Calidad en Instalaciones Nucleares en Explotación*, GS 10.7, 2000.
7. CSN, *Garantía de Calidad para la Gestión de Elementos y Servicios para Instalaciones Nucleares*, GS 10.8, 2001.
8. CSN, *Garantía de Calidad de las Aplicaciones Informáticas Relacionadas con la Seguridad de las Instalaciones Nucleares*, GS 10.9, 1998.

Asociación Española de Normalización (AENOR)

This is an entity devoted to the normalization and certification in any industrial or services area. In the catalogue of norms, part 73 is related to the nuclear industry, and series 400 to quality assurance. There is a norm

(AENOR, 1995a) of a general nature and three more specific norms (in Spanish only) devoted to design and training activities as follows:

1. AENOR, *Garantía de la Calidad en el Diseño de las Instalaciones Nucleares*, UNE 73 402, 1995.
2. AENOR, *Formación en Garantía de Calidad del Personal para Instalaciones Nucleares*, UNE 73 406, 1999.
3. AENOR, *Formación y Cualificación del Personal de Garantía de la Calidad para Instalaciones Nucleares*, UNE 73 405, 2001.

21.11 References

AENOR (1995a), *Garantía de Calidad en Instalaciones Nucleares* (in Spanish only), UNE 73 401, AENOR, Madrid.

AENOR (1995b), *Garantía de la Calidad en el Diseño de las Instalaciones Nucleares* (in Spanish only), UNE 73 402, AENOR, Madrid.

AENOR (1999), *Formación en Garantía de Calidad del Personal para Instalaciones Nucleares* (in Spanish only), UNE 73 406, AENOR, Madrid.

ANSI/ASME (2008), *Quality Assurance Requirements for Nuclear Facilities Applications*, NQA-12008, ASME, New York.

CSN (1999), *Guía Básica de Garantía de Calidad para Instalaciones Nucleares* (in Spanish only), GS 10.1, CSN, Madrid.

EFQM (2010), *Excellence Model*, EFQM, Brussels.

IAEA (1996a), *Quality Assurance in Design*, 50-SG-Q10, IAEA, Vienna.

IAEA (1996b), *Quality Assurance in Construction*, 50-SG-Q11, IAEA, Vienna.

IAEA (1996c), *Quality Assurance in Commissioning*, 50-SG-Q12, IAEA, Vienna.

IAEA (1996d), *Quality Assurance in Operation*, 50-SG-Q13, IAEA, Vienna.

IAEA (1996e), *Assessment of the Implementation of the Quality Assurance Programme*, 50-SG-Q5, IAEA, Vienna.

IAEA (2006a), *Fundamental Safety Principles*, SF-1, IAEA, Vienna.

IAEA (2006b), *The Management System for Facilities and Activities,* GS-R-3, IAEA, Vienna.

IAEA (2006c), *Application of the Management Systems for Facilities and Activities*, GS-G-3.1, IAEA, Vienna.

IAEA (2007), *Safety Glossary*, IAEA, Vienna.

IAEA (2009), *The Management System for Nuclear Installations*, GS-G-3.5, IAEA, Vienna.

INSAG (1996), *Defence in Depth in Nuclear Safety*, INSAG-10, IAEA, Vienna.

ISO (1996), *Environmental Management Systems*, ISO 14001, ISO, Geneva.

ISO (2000), *Quality Management Systems*, ISO 9001, ISO, Geneva.

KTA (1996), *General Requirements Regarding Quality Assurance*, KTA 1401, KTA – Geschaeftsstelle, Salzgitter, Germany.

21.12 Appendix: list of abbreviations and acronyms

AENOR: Asociación Española de Normalización (Spanish Association for Normalization)

ANSI: American National Standards Institute
ASME: American Society of Mechanical Engineers
CEO: Chief Executive Officer
CFR: Code of Federal Regulations
CSN: Consejo de Seguridad Nuclear (Nuclear Safety Council)
EFQM: European Foundation for Quality Management
IAEA: International Atomic Energy Agency
INSAG: International Nuclear Safety Advisory Group
ISO: International Organization for Standardization
KTA: Kerntechnischer Ausschuss (Nuclear Safety Standard)
NRC: Nuclear Regulatory Commission
OSART: Operational Safety Assessment Review Team
UNE: Una Norma Española (a Spanish norm)
USA: United States of America
WANO: World Association of Nuclear Operators

22
Commissioning of nuclear power plants (NPPs)

E. GRAUF, se-engineering GmbH, Germany

Abstract: This chapter explains the commissioning of nuclear power plants. It first highlights the importance of the commissioning for plant safety and its value for the training and qualification of power plant operating personnel. The chapter describes the roles and responsibilities of the main participants in the commissioning process, and their interactions, for example, the Operating Organization, the supplier and the Regulatory Body. The chapter also deals with typical commissioning phases, the scope of those phases and related commissioning tests. Furthermore, typical organizational arrangements, procedures and documentation needed to manage the commissioning process are explained.

Key words: nuclear commissioning, commissioning stages, commissioning manual, test procedures, test verification, pre-operational tests, first criticality, low-power tests, power tests, cold performance test, hot performance test, commissioning experience, plant handover, commissioning documentation.

22.1 Introduction

The commissioning phase is one of the most interesting and important phases in the lifetime of a nuclear power plant (NPP). It is a short but very intense period, typically encompassing 1–2 years in the total lifetime of an NPP. In no other period of the plant lifetime can operating staff gain more real in-depth knowledge and experience about plant design and plant behaviour in such a comprehensive way as is possible during plant commissioning.

The commissioning phase is essential to the subsequent safe operation of the plant, and therefore has to be carefully planned and executed. The results of commissioning have to demonstrate that the requirements and intentions of the design and the intentions of the designers, as stated in the safety analysis report, have been met and that the unit is ready for a long-lasting and successful operational phase.

In addition, with the commissioning tests the suppliers have to demonstrate that all commercial expectations fixed in the contract – such as power output, load following capabilities, availability factors, etc. – are met or are likely to be achievable within the following operating period. The commissioning phase also supports the definition of initial characteristics of systems and equipment and provides the source values for operational periodic tests.

22.2 Codes, standards and other requirements for the commissioning of nuclear power plants (NPPs)

Although the commissioning phase is extremely important in the construction of NPPs to ensure safe and reliable operation, there is very little information available in official guidelines and regulatory standards on the commissioning of NPPs. One reason might be the fact that, from the start of nuclear commissioning, all requirements for the operation, surveillance and maintenance of NPPs have to be followed, since the safety precautions at this stage do not differ from those of the normal operation phase, i.e. additional aspects to be considered in the commissioning phase are limited. Therefore, guidelines such as those of the International Atomic Energy Agency, e.g. NS-R-2, *Safety of Nuclear Power Plants: Operation* (IAEA, 2000) and others, are also valid for the nuclear commissioning phase.

Only the IAEA provides more specific guidance with its Safety Guide (IAEA, 2003) *Commissioning of Nuclear Power Plants*. Few countries have transferred the recommendations from the IAEA into dedicated national guidelines; one exception is the Finnish Regulatory Body (STUK), with its nuclear guide *The Commissioning of a Nuclear Power Plant* (STUK YVL 2.5) (STUK, 2003).

In most countries the aspect of commissioning is dealt with in other regulatory documents, for example the requirements for licence application. A typical example is provided by the UK step-by-step guide from the Health and Safety Executive (HSE), *Applying for a Nuclear Site Licence for New Nuclear Power Stations* (HSE, 2008).

However, in these cases, the aspects related to commissioning usually focus on the necessary preconditions to be provided by the Operating Organization in order to permit the nuclear commissioning. Typical preconditions include the provision of adequate documentation to demonstrate that all necessary safety arrangements, in particular adequate emergency procedures, are in place. Often these guidelines include test stages to be followed and/or hold points to allow the regulator to verify the test results before giving permission to continue the commissioning process to the next stage.

Owing to the fact that nuclear regulations in general contain no details about the scope and content of the commissioning of NPPs, it is usually the Operating Organization, in cooperation with the plant supplier, that is responsible for providing a commissioning programme for the regulator; this programme will cover the administrative arrangements, the safety precautions and the test programme, including all test procedures and success criteria – as described in the following subsections. Based on the documentation provided, the regulator may then permit the start of nuclear commissioning.

22.3 Commissioning programme and stages of commissioning

The commissioning programme of an NPP includes all the tests to verify the completion of the construction and the plant's readiness for safe operation. The commissioning programme is usually divided into stages. A review of the test results of each stage has to be completed before continuing to the next stage. Following the review, a judgement is made on whether the commissioning programme can continue to the next stage, and whether the succeeding stages need to be modified as a consequence of the test results or because some tests in the stage had not been undertaken or had not been completed.

On the basis of the broad range of commissioning practices in the nuclear industry, the commissioning process is usually divided into the following stages:

- Pre-operational tests
- Fuel loading and subcritical tests
- Initial criticality and low-power tests
- Power tests.

Often additional sub-stages are defined.

For the design of the commissioning process and its different stages, some generic aspects have to be considered:

- Each stage and sub-stage needs careful planning, preparation and description using commissioning procedures explained in Section 22.8.
- The sequence of tests within each sub-stage follows the chronological order in which they are expected to be performed.
- At the end of each commissioning phase the test results must be evaluated and reviewed before the next stage is started; before the start of initial criticality tests, low-power tests and power tests, all the tests at the previous stages must be successfully completed.
- Each stage includes the tasks necessary for the preparation of the succeeding stage and in particular the availability requirements of the systems for the succeeding stage.
- To the extent practicable, the tests should be of sufficient duration to allow the systems and components under test to reach their normal equilibrium conditions, thus reducing the probability of failure in the

early stages of operation. Furthermore, careful consideration should be given to the demonstration of the capability of the systems and components to withstand failures and/or malfunctions that previous experience has shown may occur over the expected plant lifetime.
- To ensure plant safety, all relevant safety system settings and alarm settings, including those of radiological protection instruments, need to be specified at the appropriate commissioning stages.

The design of the complete test programme and its test procedures is mainly based on suppliers' and Operating Organization know-how and experience. However, the annex to Guide NS-G-2.9 (IAEA, 2003) provides a detailed listing of typical commissioning tests structured into the different phases. Although this list is based on water reactors, it is a valuable source of information for the development of a plant-specific commissioning programme.

22.4 Pre-operational tests

A pre-operational test mainly involves the testing of components before entering cold performance tests. Before the commencement of the initial testing of any structure, system or component the following prerequisites need to be checked:

- Completion of all construction activities associated with the system or component, including quality assurance and provision of documentation to the extent necessary and practicable
- Readiness for operation: inspection for proper fabrication (including welding) and cleanness, checking of electrical and protective devices, adjustment of settings on valve torque-limiting devices, calibration of instruments, verification of operability of instrument loops and required response times, adjustment of settings for process controllers and limit switches.

In summary, the above reviews have to ensure that the construction is of the appropriate quality and that the equipment is in a fit state for commissioning to be started. In most cases these reviews are part of an organized handover process from the supplier's construction team to the supplier's commissioning team.

A satisfactory pre-operational test programme considers the proper sequence of tests of electrical systems, instrumentation systems and other service systems, such as cooling water systems and fire protection systems, in order to ensure the availability of the necessary services and safety measures for the implementation of the entire commissioning programme.

Before starting with pre-operational tests – as for all the following commissioning tests – test equipment needs to be operable and properly calibrated, and response times, of recorders for example, are in accordance with the specification described in the test procedures. Often the pre-operational stage is divided into two additional sub-stages, the cold performance tests and the hot performance tests.

22.4.1 Cold performance tests

Cold performance tests include the initial start-up of fluid systems and support systems. The objective of this stage is to obtain initial operational data on equipment, ensure compatibility of operation with interfacing systems and verify the functional performance of these systems. The tests usually include pressure testing of the primary, secondary and other supporting systems.

22.4.2 Hot performance tests

Hot performance tests are undertaken to verify that the systems conform to the specified requirements. Where possible, these tests follow cold performance tests, simulating plant operating conditions as far as is practicable, including anticipated operational occurrences at typical temperatures, pressures and flow rates.

The tests verify, to the extent possible, the effectiveness of heat insulation and heat removal systems. They enable initial checking of flow rates, vibration, clearances and other provisions made for accommodating the thermal expansion of components or systems. The operation of instruments and other equipment at high temperature is verified and the relevant operating techniques are confirmed.

The duration of hot performance testing is extended until a steady-state operating condition is achieved, in order to determine whether the structures, systems and components are operating according to specifications. It is good practice to start from the beginning of this sub-stage with the use and verification of the operating procedures.

22.5 Nuclear commissioning

22.5.1 Fuel loading and subcritical tests

With the arrival of fuel on site, and in particular the fuel loading into the reactor core, the nuclear commissioning starts; this is an important milestone in completing the project. As radiation risks will arise for the first time during nuclear commissioning and radioactive waste will be generated, it is essential that all safety provisions and emergency preparedness programmes are in place before the nuclear commissioning phase begins. The purpose of the stage of fuel loading and subcritical tests is to ensure that the fuel is loaded into the reactor safely in accordance with the loading pattern precalculated in the design. In addition, at this stage the reactor is checked to confirm whether it is in a suitable condition to be started up and that all prerequisites for permitting the reactor to go critical have been met.

The beginning of initial fuel loading is the commencement of *nuclear operation*; from this point onwards the relevant safety requirements for plant operation apply. Responsibility for meeting these safety requirements usually rests from this juncture with the plant manager. However, as explained in Section 22.7.4, sometimes other arrangements are made.

With the core loaded and the reactor maintained in a subcritical condition, a series of performance tests is carried out. These include checks on coolant flow rates, instrumentation, control rod mechanisms, automatic rod insertion and other important features of the primary circuit. Special attention is given to vibrations of core internals, fretting and loose parts signals and other phenomena that may result in component degradation.

Where necessitated by the reactor design, system flow tests and, as usual in pressurized water reactors (PWRs), cold and hot primary function tests of appropriate duration are made with the loaded core. Prior to and with the core loading itself, appropriate tests of fuel handling equipment are performed. In addition, radiological surveys and functional tests of radiation protection equipment are common practice.

22.5.2 First criticality

The highlight of any nuclear commissioning is the initiation of first criticality of the NPP. There is no doubt that all required safety provisions and arrangements for handling emergencies must be in place, and in most countries they must be verified by the regulator as a basis for his permission to start up the reactor. Fuel loading and subcritical tests have demonstrated that the core is in accordance with the design criteria and that safety systems are ready for operation. Final checks of the reactivity control systems are carried out before initiating first criticality: for example, checking of control rod withdrawal and insertion times in cold and hot conditions, checking that boron concentration in PWRs is as designed, checking the reactor protection system and protective interlocks, checking that reactivity measurements like source range and all other neutron measurements are calibrated and tested. Furthermore, shutdown margins and chemical conditions of reactor coolant and coolant within other systems have to be in accordance with the Technical Specifications of the NPP.

If all the above safety prerequisites are met and the regulator's permission is granted, the reactor operator can initiate reactor criticality depending on plant design, with the withdrawal of control rods for boiling water reactors (BWRs) and advanced gas-cooled reactors (AGRs) or the reduction of boron concentration for pressurized water reactors (PWRs). This first start-up procedure is usually supervised by the responsible plant management, supplier's commissioning personnel and reactor core physicists.

22.5.3 Low-power tests

At the stage of initial criticality and low-power tests, the initial criticality of the loaded core is achieved for the first time. The subsequent low-power tests are carried out to confirm the following:

- The performance of the reactor core is commensurate with predictions made in the core design.
- The reactor core is in a proper condition for operation at higher power levels and the characteristics of the reactor core coolant, reactivity control systems and shielding are appropriate.
- The reactor physics parameters are in accordance with predictions made in the design.

In order to permit power testing, assurance will first be obtained on the basis of the information gained from these low-power tests that there is no serious discrepancy between measured values of reactor physics parameters and other parameters and values used in the safety analysis report. The power levels at this stage are the lowest that will give reliable and stable measurements and which enable the conditions required to perform the specified tests to be achieved. Usually, special start-up instrumentation is added for this purpose and the trip limits of the nuclear flux channel for the reactor protection system are set to a conservative low level.

After achieving initial criticality, additional tests are performed as necessary to verify that the behaviour and characteristics of the core, cooling system, reactivity control systems, reactor physics parameters and shielding are as expected, and that the reactivity coefficients are as assumed in the safety analysis report. Tests are also performed to confirm the operability of plant systems and design features that could not be completely tested during the pre-operational test phase owing to the lack of an adequate heat source for the reactor coolant system and the main steam system. The following list, derived from the IAEA NS-G-2.9 Safety Guide (IAEA, 2003), is illustrative (but not a comprehensive record) of the tests to be conducted, as applicable, if they were not completed previously during pre-operational hot functional testing:

- Neutron and gamma radiation surveys
- Determination that there is an adequate overlap of source range and intermediate range neutron instrumentation, and verification of alarms

and protective functions intended for operation in the low-power test range
- Checks on changes in detector sensitivity as a result of changes in temperatures of coolant and shielding
- Comparison of the actual critical configuration with the predicted configuration
- Measurement of the temperature reactivity coefficient for poison and moderator and/or coolant over the temperature range and poison concentration range in which the reactor may become critical
- Test of scram time for control rods and shutdown rods at rated temperature in the reactor coolant system
- Determination of reactivity worth for control rods and the control rod bank, including verification of the rod insertion limits required to ensure an adequate shutdown margin, consistent with the assumptions for accidents (e.g. with the control rod of greatest reactivity worth failing to enter the core)
- Determination of the reactivity worth of the most reactive rod
- Measurements of absorber reactivity worth
- Determination of the absorber concentration at the initial allocation of criticality and reactivity
- Flux distribution measurement with normal rod patterns (this may be performed at a higher power, consistent with the sensitivity of in-core flux instrumentation)
- Operability of the control rod withdrawal and insertion sequencers and of the inhibit or block functions associated with control rod withdrawal up to the reactor power level at which such features must be operable
- Chemical and radiochemical measurements to demonstrate the design capability of the chemical control systems and of the installed analysis and alarm systems to maintain water (or gas) quality within limits in the moderator, reactor coolant and secondary coolant system
- Chemical tests of control fluid quality
- Verification of the proper response of radiation monitors to a known source
- Confirmation of the calibrations of reactivity control devices as predicted for standard rod patterns (for non-standard patterns, the differential and integral reactivity worth are to be determined)
- Leak tests of the reactor coolant system
- Measurements or checks of reactor vessel internals and of the vibration of components of reactor coolant systems
- Verification of piping and component movements, vibrations and expansions for the acceptability of safety systems
- Operability, including stroke times of isolation valves and bypass valves for the main steam line and branch steam line at rated temperature and pressure conditions

- Operability of the leakage control system for the main steam isolation valves
- Operability of the computer system for process control
- Operability of pressurizer relief valves and main steam system relief valves at rated temperature
- Operability of residual heat removal systems or decay heat removal systems, including atmospheric steam dump valves and turbine bypass valve
- Operability of purification and clean-up systems for the reactor coolant system.

22.5.4 Power tests

A comprehensive range of power tests is carried out to confirm that the plant can be operated in accordance with the design intent and that the plant can continue to be operated in a safe manner. This stage in general is limited to those tests that can be carried out only at power. Usually the commissioning proceeds with a step-by-step approach to full power and full power tests. At each sub-stage a series of tests are carried out at specified power levels. Typical steps are 10, 30, 60, 95 and 100% of full power.

The basis for the determination of tests in the power commissioning phase is those transients and events the plant is designed for, according to the Safety Analysis Report. The following list, again derived from the IAEA NS-G-2.9 Safety Guide (IAEA, 2003), is illustrative (but not a comprehensive record) of typical types of performance demonstrations, measurements and other power tests for water reactors:

- Evaluation of core performance: reactor power measurements, verification of the calibration of flux and temperature instrumentation, with sufficient measurements and evaluations conducted to establish flux distributions, local surface heat flux, linear heat rate, departure from nucleate boiling ratio, radial and axial power peaking factors, maximum average planar linear rate of generation of heat, minimum critical power ratio and quadrant power tilt throughout the permissible range of power to flow conditions
- Test of dropped rod: effectiveness of instrumentation in detecting a dropped rod and verification of associated automatic actions
- Verification of scram times after plant transients that result in scrams
- Determination of the reactivity worth of the most effective rod
- Evaluation of flux asymmetry with a single rod assembly both fully and partially inserted below the control bank, and evaluation of its effects

- Operation of control rod sequencers, reactivity worth minimizers for control rods, rod withdrawal block functions, rod runback, partial scram and 'select rod insert' features
- Operation of reactivity control systems, including functioning of control and shutdown rods and poison addition systems
- Calibration of reactivity control devices, as necessary, and verification of the performance of major or principal plant control systems such as the average temperature controller, automatic reactor control systems, integrated control system, pressurizer control system, reactor coolant flow control system, main, auxiliary and emergency feedwater control systems, hot well level control systems, steam pressure control systems and reactor coolant make-up and let-down control system
- Measurement of power control by flow variation and demonstration of flow control
- Rod pattern exchange demonstration (at the maximum power that rod exchange will be permitted during operation)
- On-power refuelling tests if applicable
- Tests of power reactivity coefficients or power versus flow characteristics
- Tests of dynamic plant response to the design load following operation, including step and ramp changes, and response to automatic control
- Test of the dynamic response of the core and plant to fast load changes initiated by the load control
- Test of the capability of plant systems to control oscillations in xenon levels in the core
- Operation of the reactor coolant system with the plant in steady-state condition to establish flow rates, reverse flows through idle loops or jet pumps, core and channel flow, differential pressures across the core and major components in the reactor coolant system, and vibration levels of other components
- Natural circulation tests of the reactor coolant system
- Determination of baseline data for the monitoring system for loose parts of the reactor coolant system
- Vibration monitoring of reactor internals in steady-state and transient operation, if this testing has not been completed previously
- Demonstration of effectiveness of leak detection systems for reactor coolant, if not previously demonstrated
- Operation of failed fuel detection systems in accordance with predictions
- Radiation surveys to determine the effectiveness of the shielding
- Process radiation monitoring systems and effluent radiation monitoring systems: correctness of response
- Chemical analyses (at frequent intervals)

- Functioning of chemical and radiochemical control systems and sampling to verify that the characteristics of the reactor coolant system and secondary coolant system are within specified limits
- Operation of processing, storage and release systems for gaseous and liquid radioactive wastes
- Effluent monitoring systems: verification of calibration by laboratory analysis of samples (as early in power ascension as possible and repeated at defined power steps)
- Process computer – comparison of safety-related predicted values with measured values; verification of inputs to control room computers or process computers from process variables, data printouts and validation of performance calculations performed by the computer; validation of all computer safety functions
- Turbine trip tests
- Tests of generator main breaker trip, with the method used for opening the generator output breakers (by simulating an automatic trip) selected such that the turbine-generator set will be subjected to the maximum credible overspeed condition they could encounter during plant operations
- Tests with loss of off-site power up to 100% of generator power output
- Functional tests of relief valves – verification of operability, response times, set points and reset pressures, as appropriate, for pressurizer relief valves, main steam line relief valves and atmospheric steam dump valves
- Verification of operability and response times of isolation valves for the main steam line and the branch steam line
- Evaluation of performance of shutdown cooling system – capability of all systems and components provided to remove residual heat or decay heat from the reactor coolant system, including condenser steam dump valves or atmospheric steam dump valves, the residual heat removal system in steam condensing mode and the reactor core isolation cooling system, and testing of the auxiliary feedwater system to include provisions that will provide reasonable assurance that excessive flow instabilities (such as 'water hammer') will not occur during subsequent normal system start-up and operation (before exceeding 25% power)
- Determination of the dynamic response of the plant and the subsequent steady state of the plant for single and credible multiple trips of the reactor coolant pumps (PWRs) or the circulator and/or failure of the flow control valves of the reactor coolant system
- Trip of feedwater pump and restart of standby pump
- Test of the dynamic response of the plant for a simulated condition of loss of turbine generator coincident with loss of off-site power
- Test of the dynamic response of the plant to load rejections, including turbine trip

- Test of the dynamic response of the plant for the case of automatic closure of all main steam line isolation valves (for PWRs the test may be made at a lower power level to demonstrate proper plant response to this transient)
- Test of the dynamic response of the plant to loss or bypassing of the feedwater heater(s) due to a credible single failure or operator error that results in the most severe case of a reduction in feedwater temperature
- Observations and measurements, as appropriate, to ensure that piping and component movements, vibrations and expansions are acceptable for safety systems (tests performed in low power testing need not be repeated)
- Performance of the auxiliary systems whose operable components have the minimum design capability for operation of the engineered safety features
- Tests of the load-carrying capabilities of systems, components and cables
- Performance of ventilation systems and air conditioning systems
- Test of shutdown from outside the control room and plant operation from the emergency control room according to the plant safety design.

All the tests are designed to demonstrate to the extent practicable that the plant operates in accordance with the design in steady-state conditions and load-follow operation, during and after anticipated operational occurrences, including reactor trips and load rejections initiated at appropriate power levels.

Often the power test phase ends with a *Commercial Power Operation Acceptance Test* lasting 2–4 weeks or even more on full power and/or load-follow operation according to the contractual arrangements between the Operating Organization and the supplier. If successfully completed, the plant is handed over to the Operating Organization.

At the end of all commissioning tests it is recommended to carry out a final review to confirm whether the operational limits and conditions are adequate and practicable, and to identify any constraints on the operation of the plant that the commissioning tests have shown to be necessary.

22.6 Roles and responsibilities during commissioning

Organizational arrangements are necessary to achieve the safety objectives of commissioning and nuclear safety in general. They determine exactly the roles and responsibilities of all parties and functions involved in the commissioning, particularly the safety-relevant roles. Furthermore, they represent a convenient and practical working scheme which allows optimum use of available personnel, materials and methods, and which enables assurances on safety to be obtained. The organizational arrangements have to include all of the parties involved in the commissioning process, in particular the main players: the Operating Organization, the plant designer/supplier, various support organizations and last, but not least, the Regulatory Body.

22.6.1 Operating Organization

Although responsibility for commissioning activities may be assigned to a contractor or the construction organization, the Operating Organization is fully responsible for the commissioning of its nuclear installation. In some cases responsibility may be transferred from one organization to another at the time of fuel loading or at some other appropriate hold point. However, whatever the arrangement, the organization or individual responsible for commissioning should be accountable to the organization or individual responsible for compliance with the licence for the following:

- Demonstrating that the plant behaves in accordance with the design intent
- Confirmation that the plant has been tested within the design limits only
- Ensuring that safety requirements are observed while the commissioning process is being conducted.

To fulfil these general requirements the responsibilities of the Operating Organization are supported by administrative measures:

- To control, review and coordinate the activities of the construction, commissioning and operating groups in an effective manner
- To ensure that the commissioning procedures are prepared, reviewed and approved by personnel with appropriate technical backgrounds and experience
- To arrange for the required submissions to the Regulatory Body at the approved stages and to comply with its requirements
- To establish procedures for ensuring the coordination of commissioning activities, account being taken of the views and experience of members of the construction, commissioning and operating groups as well as other participants such as those from the designers, the manufacturers and the consultants and quality assurance personnel
- To ensure the maintenance of adequate numbers of properly trained, experienced, qualified and, where required, authorized personnel in the commissioning and operating groups
- To receive and disseminate the requirements of and information from the Regulatory Body.

Some Operating Organizations have installed a nuclear safety office or senior advisory group to evaluate safety issues that is independent of the plant/commissioning manager and that has direct access to the top management of the licensee organization.

22.6.2 Supplier

In many cases the supplier takes the lead in designing the commissioning process, its tests and execution, although the Operating Organization holds overall responsibility for nuclear safety and operates the plant. In these cases the commissioning manager is usually employed by the supplier. Such kinds of arrangement need exact definition of the roles and responsibilities within the *Commissioning Manual* (see also Section 22.8.1). In particular, the arrangements have to ensure that there are clear information lines from the commissioning manager to the responsible management of the Operating Organization. Whatever the organizational arrangements for the commissioning of the nuclear power plant are, the Operating Organization has to review and to approve the commissioning programme and the related test procedures.

The supplier – particularly if the commissioning is part of his contractual obligations – supports the commissioning with his own commissioning group but, in any case, with specialist knowledge, expertise and relevant experience from plants already commissioned. Other obligations are the provision of all relevant information and baseline data, to assist in the analysis of discrepancies and unexpected events. If necessary and required by the test results, the supplier has to rectify design deficiencies and to provide complete documentation of the modification carried out, including requalification of test results.

22.6.3 Support organizations

The responsibilities of other participants, such as designers, manufacturers and supporting technical organizations in the commissioning activities need to be specified in the related contracts and in the Commissioning Manual.

22.6.4 Regulatory Body

All of the activities in the commissioning phase are subject to surveillance by the Regulatory Body, which is also responsible for granting authorization for the commissioning activities and operation. The Operating Organization will have to request such an authorization. A complete set of safety documents, typically including a final safety analysis report, the technical specifications for operation, the radiation protection manual, the emergency plan, and emergency and routine operating procedures, the quality assurance programme for operation and the surveillance test programme are requested from the Operating Organization to be reviewed by the Regulatory Body prior to the start of nuclear commissioning.

For the commissioning process itself the Regulatory Body reviews and approves the commissioning programme as required by national practice. Hold points are usually established in the commissioning programme in order to assess test results before regulatory authorization is given to proceed. Before authorizing the loading of nuclear fuel or initial criticality, the Regulatory Body has to complete as appropriate the review and assessment of aspects such as the as-built design of the plant, the results of pre-operational tests, and the adequacy of operating procedures and instructions, especially main administrative procedures, normal operating procedures and emergency operating procedures (EOPs). Another important aspect to be reviewed is the fulfilment of staffing and qualification requirements and confirming that corresponding training is completed as required.

Prior to granting permission for acceptance of fuel on site, the measures for accounting fissile and radioactive materials and the fulfilment of the applicable requirements in respect of safeguards must be completed and verified by the Regulatory Body. In addition, all technical and administrative procedures for the physical protection of the site need to be in place and accepted by the regulator. Before licensing and/or authorizing routine operation at full power, the Regulatory Body has to complete the review and assessment of the results of commissioning tests and their analysis, in addition to other aspects that need to be reviewed in this regard.

22.7 Commissioning organization and management

The principal activities performed in commissioning can be divided into three categories:

1. Those associated with the final stage of construction and installation of the plant
2. Those specific to commissioning, including safety reviews
3. Those associated with the operation of the plant.

Accordingly, personnel performing the above activities belong to the following groups:

- Construction group
- Commissioning group
- Operating group.

In addition, there are other representatives participating in commissioning activities, such as representatives of the designers, the construction group, the component manufacturers and the Regulatory Body. These representatives collaborate with the aforementioned groups as appropriate. In particular, the designers and manufacturers provide adequate and complete information to the groups. It is good practice, sometimes requested by the Regulatory Body, that the design engineers are involved in the review of commissioning data to confirm that the performance meets the design intent.

There are various ways in which the construction, commissioning and operating groups could be formed by different organizations. This may depend on the industrial practice and experience of nuclear power in the State or country, on the supplier and the Operating Organization, on contractual arrangements, as well as on the physical size and design of the plant. The composition of the groups may also be influenced by the availability and experience of personnel performing specialized functions. In most cases the Operating Organization takes over the responsibility for nuclear safety and plant operation during commissioning, usually from core loading or first fuel on site.

In some cases, the Operating Organization decides to contract the commissioning activities to another organization, e.g. the supplier, as is the rule in turnkey contracts. However, in these cases it also has to be made clear that the ultimate responsibility for safety cannot be delegated and remains with the Operating Organization. Generally, the construction group is responsible for ensuring that the installation has been completed in accordance with specifications. The commissioning group has to ensure that structures, systems and components are tested to provide assurance that the plant has been properly designed and constructed and is ready for safe operation. The operating group has to operate systems and plant in accordance with the assumptions and intent of the commissioning programme, and last but not least in accordance with the safety requirements, in particular the *Limits and Conditions for Safe Operation* (LCOs).

Since construction, commissioning and operating activities overlap, in the arrangements made in respect of utilization of personnel among the construction, commissioning and operating groups, it is essential that responsibilities remain clear at all times. It is highly recommended that working arrangements, as far as practicable, make use of the operating personnel so that they become familiar with the plant and the facilities during commissioning. As a consequence, the operating group has to take all necessary actions to have suitable trained operating staff members ready from the beginning of the commissioning activities in order to ensure that as many operating personnel as possible gain field experience and to establish an 'institutional memory' of the plant. The human resource development plan and the training programme, in particular for the engineering resources, the operators and the maintenance personnel, have to consider this essential requirement (see also Section 22.10).

22.7.1 Organization of the commissioning group

The commissioning group and the special arrangements made to ensure proper coordination of commissioning activities need to be established early enough to allow all these activities to be identified and adequate preparations to be made. As mentioned previously, all organizational arrangements, processes and procedures relevant for the commissioning phase are part of and are described in a Commissioning Manual. The Commissioning Manual – explained in more detail in Section 22.8.1 – needs to be integrated and well synchronized with the overall plant management system of the Operating Organization, as well as with other systems such as the supplier's management system, to prevent conflicting rules and regulations.

The commissioning group is headed by a commissioning manager who has had sound experience with NPPs. Depending on organizational and contractual arrangements, the commissioning manager usually belongs to the plant operator's or to the supplier's organization. Whatever the arrangement, the commissioning manager needs to be appointed well in advance of the actual commissioning work so as to be able to make the necessary arrangements for scheduling and organizing work units, work plans and other resources.

Specific test teams are in charge of performing commissioning tests. The number and composition of these teams depends on matters such as:

- The number and complexity of the systems to be tested
- The workload
- The skills necessary to perform the tests
- The staff available from either the supplier or the Operating Organization.

Each test team is usually led by a test team leader with appropriate experience in the operation or commissioning of NPPs.

A planning and scheduling unit is necessary in the commissioning group to develop commissioning schedules, to monitor and to report on the progress of commissioning in all its aspects, including the issuing of commissioning reports. The responsibilities of the commissioning group include the following:

- To plan in advance the commissioning programme with detailed test sequences, time schedules and staffing requirements
- To update the commissioning programme in the light of experience in commissioning and as a result of design modifications
- To establish a procedure for the preparation, review and approval of test procedures and other procedures

- To ensure that operational flow sheets, operating and maintenance instructions, commissioning procedures, formats for commissioning reports and test reports, plant handover documents and submissions to the Regulatory Body are available
- To establish a procedure for the systematic recording of plant data for future use
- To establish a procedure for ensuring that incidents in commissioning are analysed so that the experience gained can be fed back to the designers or the operating group
- To verify that the installation of structures, systems and components has been satisfactorily completed and codified for proper identification
- To ensure that the prerequisites for the commissioning programme have been satisfied and that pre-operational tests such as functional checks, logic checks, interlock checks and system integrity checks have been completed
- To ensure that the commissioning procedures comply with the appropriate rules and regulations for safety (including radiological protection and nuclear safety)
- To ensure that the systems are commissioned safely and to confirm that the written operating procedures are adequate
- To implement all the tests in the commissioning programme, including repeat testing of the systems that have been commissioned initially as partially installed
- To make suitable arrangements for testing and maintaining systems (particularly safety-related items) for which responsibility has been accepted
- To direct the operation of systems in the commissioning programme and to update operational flow sheets and operating and maintenance instructions, as well as procedures based on experience in commissioning
- To establish procedures for analysing the results of tests and for producing test reports and test certificates
- To issue commissioning reports on tests
- To ensure that a procedure is in place to control the calibration of test and measurement equipment
- To establish a procedure to ensure that all participants in the commissioning process are suitably qualified and experienced
- To ensure the configuration management, maintaining consistency between 'as built' drawings and procedures and physical configuration and design requirements
- To ensure that design changes are requested, reviewed and implemented when design criteria are not met or when they fall short

- To establish a procedure for controlling temporary changes to plant and equipment
- To issue test certificates and stage completion certificates or their equivalent
- To provide up-to-date baseline information to the operating group and the Operating Organization
- To report to the Operating Organization any deficiency detected in commissioning tests in order that corrective action can be taken
- To maintain a record of limiting conditions in commissioning
- To ensure that plant performance is in accordance with the design intent, including all aspects of radiological protection and safety
- To certify that the commissioning programme has been satisfactorily completed
- To establish and implement procedures that ensure the orderly transfer of responsibilities for structures, systems and components from the construction group to the commissioning group, and from the commissioning group to the operating group, using a system of documents such as transfer certificates
- To ensure that an opportunity is provided for operating personnel to gain plant experience, typically by utilizing the appropriate personnel, as necessary, for commissioning activities.

22.7.2 Organization of the operation group

The Operating Organization, as the organization authorized by the Regulatory Body to operate the plant, acts as the overall controlling and coordinating authority for overseeing the safe and satisfactory completion of all commissioning work. When commissioning activities are conducted under the responsibility of the contractors, the Operating Organization has to implement necessary arrangements to review and approve these activities at all stages. In a more practical sense, the operating group, as a part of the Operating Organization, is usually one of the key players in the commissioning process. The operating group is usually composed of staff from the plant operation department. Shift personnel form the main body of the operating group. The operating group operates the plant from the main control room (MCR) with the control room shift supervisor and related desk operators, and in the field with the field operators.

The responsibilities of the operating group within the commissioning process are as follows:

- To carry out operation and maintenance with competent staff to meet the needs of the commissioning programme
- To participate in the commissioning activities

- To take responsibility for the systems completely commissioned and transferred to the operating group
- To satisfy the Operating Organization that the systems that are transferred comply with specified performance requirements, the design intent and safety requirements
- To become competent in the methods of operation of the plant
- To verify the operation procedures in terms of correctness and practicability.

The Operating Organization has to take all necessary actions to enable the operating personnel to fully participate in commissioning activities at the plant at all levels, thus providing the operating staff with an opportunity to become familiar with and gain experience of the plant. This approach to training and preparation of the operating staff during commissioning will contribute towards the assurance of safety during the initial operation of the plant.

Procedures for operating and periodic testing should be used as far as the conditions of the plant will allow in the commissioning phase so as to validate them prior to initial loading of the core. Personnel should also adhere to normal operating rules as far as applicable, such as those relating to access to the control room, control of information, control cabinets and switchboards, communications with the control room about abnormalities and changes in plant configuration. The need for adherence to normal operating rules should be re-emphasized to personnel after the core has been loaded.

22.7.3 Interfaces between participants in the commissioning process

Many activities are performed in parallel with the commissioning of the plant, such as those relating to construction, operation and maintenance. The interfaces between these activities have to be adequately managed to ensure the protection and safety of the plant and personnel and to ensure that the commissioning programme is not impaired. Appropriate work control processes have to ensure the proper coordination of all group activities involved in commissioning and to cover the major work activities, including post-work testing. This process also provides for the proper channelling of the work to the person responsible for the system and for ensuring notification and awareness in the control room of all the work activities that are in progress.

Interface between construction activities and commissioning activities

Clear and well-understood lines of authorization and communication between construction and commissioning activities must be established and documented so as to manage a rigorous work prioritization policy. Clear lines of communication support the commissioning schedule and the agreements on the scope of activities in both organizations, in particular at the interfaces.

Since the construction organization has responsibility for certain activities during the commissioning programme, the different roles and responsibilities should be defined well in advance of the commencement of this programme in order to prevent misunderstandings. Particular areas of consideration for the interface are procedures for transferring structures, systems and components from construction to commissioning. Special precautions are also necessary for the commissioning of partly installed systems.

Interface between commissioning activities and operating activities

As with the interface between the construction and the commissioning groups, the interface between the commissioning and the operating groups needs detailed rules and regulations. The following particular aspects are of relevance:

- Procedures for transferring structures, systems and components for operation
- Methods of identifying the special technical, operational or staffing restrictions necessary as a result of partial completion of a construction or commissioning activity
- Changes in responsibility for safety including the nomination of responsible persons
- Modifications to the plant and to the procedures
- Availability of as-built drawings, instructions and procedures for operating and maintaining the systems and the plant
- Control of temporary procedures and equipment available during commissioning but not appropriate for normal operation, for example, special start-up instrumentation or duplicate safety keys and authorization for the use of jumpers and lifted leads
- Provision of sufficient opportunity for the operating personnel to become both trained in and familiar with the operating and maintenance techniques for the plant.

Surveillance and maintenance during commissioning

From construction to commissioning and finally to operation, the plant must be adequately monitored and maintained. It should be subject to the required periodic tests and inspections in order to protect equipment, to support the testing phase and to continue to comply with the safety analysis report. Historical records of operation and maintenance are kept from the time of initial energization and operation of each plant system, and transferred with the documentation package to the Operating Organization (see Section 22.13).

The organization for surveillance and maintenance during commissioning, and related roles and responsibilities, must be adequately described and documented so as to be clear to all the parties involved. In particular, the scope of the responsibilities of the construction and operating groups in relation to surveillance and maintenance during commissioning needs to be clearly identified. The organization established for maintenance during commissioning has to ensure that the maintenance group of the Operating Organization becomes actively involved at all levels in the organization for maintenance during commissioning. The participation of personnel from the instrumentation and control section in particular should be encouraged. Recommendations and guidance on maintenance activities can be found in the IAEA Safety Guide NS-G-2.6 (IAEA, 2002).

22.7.4 Transfer of the plant from the supplier to the operating organization

Plant handover is the transfer of responsibilities for the plant. This includes structures, systems and components, items of equipment and documentation, and may include personnel. There are various concepts and related contractual arrangements for plant handover. Two models are mainly used for this purpose.

1. With turnkey contracts the complete plant is handed over to the Operating Organization after the successful completion of all commissioning tests, including the *Commercial Power Operation Acceptance Test*. From a contractual and organizational point of view this is the most transparent and easy to handle solution. In some cases, a step-by-step handover approach is selected even in turnkey contracts; here the handover procedures and the graded transfer of responsibilities are more complicated and require very detailed arrangements.
2. Often, alternatives to turnkey contracts have been selected by the Operating Organization and plant handover starts with the commissioning activities or at the latest stage with nuclear commissioning. As with the graded approach in turnkey contracts, the arrangements related to the handover, and even more those related to the responsibilities of the various partners, require very detailed study and definition.

The most important transfer of responsibility is the transfer of responsibility for nuclear safety. Special care should be taken to ensure that responsibilities for personnel, plant and safety are clearly defined and rest with the appropriate organization. It is the Operating Organization's responsibility to ensure that an appropriate procedure for the handover of the plant is in place, describing detailed steps in the handover process, including responsibilities and authorities of the parties involved. However, it is highlighted here again that it is best practice and in accordance with international nuclear standards that from the time of the arrival of nuclear fuel at the site, responsibility for safety rests with the Operating Organization.

The transfer of documentation is another key aspect within the plant handover process and needs special attention. More details are provided in Section 22.13.

22.8 Commissioning procedures

A commissioning programme has to identify and describe all the tests and related activities necessary to demonstrate that the plant has been properly designed and constructed and can be operated safely. The commissioning programme should be written in such a form as to enable the objectives and methods of testing to be readily understood by all concerned and to allow control and coordination by management. For multi-unit plants a separate programme is produced for each unit. It is good practice to collect all relevant administrative and technical procedures related to the commissioning programme in a comprehensive document such as a Commissioning Manual.

22.8.1 Commissioning Manual

In order to ensure an effective and safe execution of the commissioning process, activities and measures have to be carefully defined and established in written procedures. A clear definition of tasks, responsibilities and interfaces between the entities involved in the commissioning activities is also required. All these procedures constitute the Commissioning Manual, which should define the following:

- The role and the responsibilities of every entity involved in the commissioning activities
- The tests and the applicable processes and workflows including their interfaces for the performance of commissioning
- The technical conditions for the performance of commissioning activities
- The structure of the technical documentation to be used to fulfil the commissioning activities, including reporting requirements.

The Commissioning Manual usually consists of an organizational part and a technical part. Typical topics within the organizational part are:

- Commissioning organization
- Commissioning system transfer process

- Commissioning tests performance
- Handling of modifications and deviations
- Commissioning documentation management
- Quality and environment
- Health and safety during commissioning.

The technical part of the Commissioning Manual is constituted by a set of commissioning programmes, instructions and worksheets, for example:

- A general commissioning programme describing the different phases of commissioning activities
- A phase-oriented commissioning programme for the whole plant related to a particular commissioning phase, listing all the activities (including operating ones) to be performed for the whole plant during this commissioning phase, and also including the prior conditions for the starting of the overall commissioning phase concerned, as well as all waivers required with respect to the LCOs for performance of some tests after fuel loading
- A system-oriented commissioning programme related to a system (or group of systems)
- All the tests that are necessary for proving the safety and the performance of the plant and the logical sequence of those tests
- Test procedures for each of the specified tests. More details about test procedures are provided in Section 22.9.

Whatever the organizational arrangements for commissioning of an NPP, the Operating Organization has to review and approve the Commissioning Manual; furthermore, it is common practice that the commissioning programme is also submitted to the Regulatory Body for review and approval.

22.8.2 Commissioning programme

Commissioning is essential to the subsequent safe operation of the plant and therefore needs to be carefully planned and executed. The commissioning covers all the activities to be performed on structures, systems and components to bring them to an operating mode. Commissioning is part of the process of verification that the provisions of the design basis are met and that the assumptions made in the safety analysis report are justified.

To fulfil these demands, an appropriate and detailed commissioning programme has to be designed. It includes tests of different types, and distinctions should be made between:

- Tests that aim at the verification of each functional system, including its overall performance
- Tests on new types of equipment

- Tests performed on the prototype plant for a series in order to test the validity of a new concept; subsequent tests on the plants in the series would then just test for conformity
- Tests aimed at acquiring data to validate the code used for the design and to confirm the validity of the limiting safety system settings
- Tests to validate operating procedures.

The programme is divided into stages whose number and size will depend upon safety requirements and technical and administrative requirements (see Section 22.3). The programme shows the planned duration of the activities and their interrelationships, and includes activities that may be necessary in order to provide opportunities for the operating personnel to gain familiarity with the operation of the plant.

The commissioning programme is structured so as to ensure that the following objectives are met:

- All the tests necessary to demonstrate that the installed plant meets the design intent stated in the safety analysis report are performed.
- The tests are performed in a systematic sequence – in particular, tests should be arranged to be progressive, so that the plant is exposed to less onerous conditions before more onerous ones.
- The programme provides means of identifying hold points in the commissioning process.
- Operating personnel are trained and procedures are validated.

The programme also includes:

- The situations at which reviews and hold points are required
- Any applicable requirements of the Regulatory Body, including the witnessing of specified tests
- The title of each test together with a unique identification
- Cross-references to other documents relevant to commissioning
- Provision for data collection for further use.

During commissioning, normal operating procedures, including those for operational periodic tests, are used as far as possible to validate the applicability of these procedures. The EOPs, which are not used in routine commissioning operations, should also be validated in the commissioning programme, as far as possible.

Testing, as the core of the commissioning programme, needs to be sufficiently comprehensive to establish that the plant can operate in all modes for which it has been designed to operate. However, tests should never be conducted, and operating modes or plant configurations should not be established, if they have not been analysed, if they fall outside the range of assumptions made in analysing postulated accidents in the safety analysis report, or if they might damage the plant or jeopardize safety.

For identical units in a multi-unit NPP and/or for a series of identical plants, it is common practice to omit selected tests that have already been performed for the units tested previously. The Operating Organization has to ensure that such an action does not jeopardize safety and that it is taken only with the prior approval of the Regulatory Body.

Special provision has to be made to ensure that the safety of another nuclear unit or other facilities close by and already in operation is not jeopardized in the commissioning tests. Such provisions include conducting a hazard assessment and obtaining the prior approval of the Regulatory Body and specific written approval from the manager responsible for the operating unit. Close liaison between the Regulatory Body and the Operating Organization throughout the development and implementation of the whole commissioning programme is recommended so as not to delay the commissioning process.

22.9 Test procedures

All commissioning tests are based on authorized written procedures. The preparation of test procedures, including their verification and approval, is carried out according to administrative procedures defined in the Commissioning Manual. The level of review depends on the importance to the safety of the system and the nature of the test.

The specific procedures are specified in the commissioning programme and are usually part of the Commissioning Manual. They describe the principles, objectives and nature of the tests. They include also the criteria for judging the validity of the results and the acceptance criteria. The procedures for systems that are important for safety contain checks that all performance levels and operating parameters have been demonstrated for all the operating configurations (normal, transient and accident conditions). Any test procedure defines in detail how each item of equipment, system or component will be commissioned, and therefore test procedures thus form the core of the commissioning process. Competent personnel and adequate controls are therefore necessary to ensure that the test procedures are of a high standard. Designers and other specialists are usually involved in formulating tests.

All procedures are subject to a thorough verification and approval process in which the regulatory authorities and the Operating Organization participate. No test procedures should be used if not verified and approved as required. The designers – even if not in charge of developing the test procedures – also have to participate in the approval process, in particular in reviewing the validity of the acceptance criteria.

The test procedures should follow normal plant operating procedures to the extent practicable, in order to verify them. If necessary, the normal operating procedures can be amended for use during commissioning. This helps the operating personnel to become familiar with them.

The procedures include any necessary deviations from the design or normal operating configurations, although consideration should be given to minimizing the use of such arrangements. The descriptions are sufficiently detailed to ensure that such deviations are made correctly before the start of the tests and to ensure that the systems and components are restored to their normal status once the testing has been completed. For this purpose, special arrangements such as temporary interlock bypasses, temporary additional interlocks, temporary system bypasses, valve configurations and instrument settings are identified, and the points in the test procedure and/or commissioning programme for terminating these temporary arrangements are specified.

Test procedures typically include the following:

- Identification coding and cross-references
- Introduction
- Test objectives and methods
- Limiting criteria, in particular applicable limits and conditions for operation
- Prerequisites and initial conditions
- Test conditions and procedures
- Acceptance criteria
- List of required tools, test equipment and instrumentation
- Staffing, responsibilities and qualification requirements
- Special precautions
- Completion criteria for the test
- Records, data collection and processing.

22.10 Qualification requirements for commissioning personnel and other human factors

22.10.1 Typical functions and related qualification requirements

Personnel engaged in commissioning activities must be suitably qualified and experienced for the level of responsibility and importance to safety of their work. The necessary level of qualification and experience needs to be specified for each position in the commissioning organization. In principle, from the start of nuclear commissioning, the qualification requirements should not deviate from those required for the operation of NPPs. Therefore guidelines and regulatory requirements in the field of training and qualification of NPP personnel are widely applicable to the commissioning personnel with roles and responsibilities comparable to those for normal plant operation. So, for example, commissioning managers should have comparable qualifications and experience to plant managers. More information about training and qualification of NPP personnel can be found in the IAEA requirements document NS-R-2 (IAEA, 2000) and in the technical document TRS-380 (IAEA, 1996).

The training programme for commissioning personnel, in particular the commissioning engineers and test group leaders, should cover some specific aspects relevant to commissioning:

- Methods of and techniques for commissioning
- The conducting of tests and maintaining the plant in safe conditions
- Interfaces of construction, design and operation with commissioning
- Procedural changes and design changes
- Permanent and temporary modifications.

First-line managers from suppliers and main contractors involved in commissioning activities often participate in the training programme as appropriate because of their close interaction during this phase.

During the training and day-to-day activities, a safety culture and concern for quality have to be established at all levels among the personnel involved from the early stages of commissioning (see Section 22.11). The importance of the work of those personnel performing commissioning activities, with respect to achieving quality objectives and safety objectives, should be highlighted in the training programme.

If any major incidents occur during commissioning, a root cause analysis should be carried out and any necessary improvements in training should be identified. Experience gained in commissioning has to be appropriately incorporated into the training material. The IAEA has provided guidance in this regard in technical document TECDOC-1600 (IAEA, 2008).

22.10.2 Value of commissioning experience

Another aspect related to training, in particular the training of operation and maintenance personnel belonging to the Operating Organization, is the value of commissioning experience. No other period in the lifetime of an NPP is more intense and effective for personnel training of the Operating Organization. Therefore all necessary actions should be taken by the Operating Organization to ensure that the operating personnel and engineering and maintenance staff have completed most if not all basic training prior to the start of commissioning activities to enable them to participate fully in the commissioning. In addition, provision should be made for training personnel, for example simulators or maintenance instructors, to participate in the commissioning process in certain aspects of the plant, mainly those related to design, methods of working and operation.

22.11 Safety management and development of a safety culture

Worldwide operating experience shows that with the commissioning an operational and safety culture develops. Behaviours, in particular misbehaviours, developed during commissioning and the early operational phase will last for long periods, often throughout the whole lifetime of the plant, and if corrections are needed it is rather difficult to make these changes in a later phase. As a consequence, one of the basic functions of the Operating Organization in this period is to promote and foster the development of a safety and operational culture at the plant and to ensure that attributes such as personal dedication, safety consciousness, conservative decision making and a questioning attitude become habitual in the subsequent operational stage.

Specific consideration should therefore be given to the arrangements used by the organizations participating in the commissioning process for the management of safety in order to enhance safety culture and achieve good safety performance. Safety management should be embedded in the *Integrated Management System* (IMS) of the Operating Organization and practised in all of the commissioning activities.

22.12 Recording and analysis of tests

All testing results need a thorough review. The purpose of the review is to provide assurances that the testing performed demonstrates that the performance of the systems tested is in accordance with the design intent and that any operating constraints have been identified. It should ensure that all necessary data have been obtained and analysed, and that the technical evaluation and test report have been completed. The review also provides assurances that the succeeding stages can be conducted safely and that the safety of the plant is never dependent on the performance of untested structures, systems or components. The evaluation of the test results requires a comparison with the acceptance criteria and should be independently carried out by the commissioning group, the designer and the regulator. The objective is to clarify whether the design intent has been met. Personnel assigned to carry out reviews must have adequate experience in their individual specializations – the main reason why the involvement of design engineers is highly recommended. Technical support organizations or consultants may be used for dealing with particular problems.

At the end of a stage, the results of the tests in that stage and the general condition of the plant are reviewed by the representatives of the commissioning group and the Operating Organization prior to approval being granted to begin the next stage. Depending on national regulatory practices, the Regulatory Body may be involved in the review and approval of the results of a specific stage. Progress to the next stage will be permitted by the Operating Organization when the completed review of the current stage has been approved by the Operating Organization in accordance with the requirements of the Regulatory Body.

22.13 Documentation

The whole commissioning process needs to be well documented. Documents are prepared and issued during the progress of the commissioning activities in order to certify the performance of the tests and to provide the required authorizations for the continuation of the programme, in accordance with the procedures established by the Operating Organization. Typical documentations in commissioning are test certificates and stage completion certificates. With the test certificate the result of an individual test is documented and the fact that the test has been completed in accordance with the success criteria; alternatively, any reservations or departures from the expected results are recorded. In addition, stage completion certificates are produced to record that all the tests for a particular stage have been successfully completed. All test reports and test certificates for the stage have to be completed for further review.

In particular, when the commissioning process works well, the preparation and approval of test documentation is a challenging activity if the commissioning programme is to proceed in an orderly and timely manner. Suitable preparations have to be made so that the stage completion and approval documents can be produced expeditiously. To this end, reviews of test results need to be undertaken and test results need to be accepted at suitable times during the progress of testing within each stage. The end of each stage should include preparations for the start of the succeeding stage and a means should be arranged for the continual updating of documentation. In addition, close liaison should be maintained with all participants in the commissioning programme, including personnel at the headquarters of the Operating Organization and personnel of the Regulatory Body.

Parallel to the commissioning and the related documentation activities, the plant documentation required as part of the plant handover process is also prepared. The plant documentation is usually transferred in system packages and over a reasonable period of time in order to allow the plant personnel to review each package comprehensively. This review often forms part of the operating group's activities during the commissioning process.

The following documentation is typical for a system acceptance package:

- General correspondence and system records
- Results of load tests and pressure tests, flushing records and cleaning records
- Acceptance packages from the construction, including non-destructive testing (NDT) and other welding inspection records
- As-built diagrams, electrical diagrams, instrumentation and control diagrams, and flow diagrams
- Documentation of pre-nuclear test procedures and report data sheets
- Failure reports and incident reports
- Documentation on temporary modifications, lifted leads and jumpers, and software modifications
- Equipment isolation records and work permit records
- Records of preventive and corrective maintenance
- Surveillance records
- Records of field changes and design changes
- Pending item lists including defects, omissions and weaknesses carried forward from the previous handover
- Suppliers' and manufacturers' manuals.

In performing the handover review, plant walk-downs should be carried out by representatives of the organizations involved in the handover process. Such walk-downs can be combined with the walk-downs required by the commissioning procedures.

22.13.1 Reporting

Comprehensive reports about the test results, in addition to the detailed test documentation, are common practice for NPP commissioning. Stage test reports and a final station commissioning report are prepared to satisfy the information needs of individual stakeholders, for example the Regulatory Body and the top management of the Operating Organization and the supplier.

22.14 International experience

More than 400 nuclear installations have been commissioned to date worldwide. In most cases the commissioning went smoothly without major problems, particularly when the commissioning was well prepared by the Operating Organization, the supplier and others involved. In some cases the process was delayed due to technical problems related to the plant design or due to quality problems with components. Such cases cannot be attributed to the commissioning process itself, but to the design and construction phase of the project.

In some other cases the commissioning was delayed due to non-availability of the required staff, either from the supplier or from the Operating Organization. In these cases the Operating Organization has to bear the prime responsibility for the delays or the poor performance of the plant commissioning.

Good communication and cooperation with regulatory bodies and the main contributors, such as the suppliers and manufacturers, prior to and during commissioning is recognized as one of the key success factors for an effective commissioning process and, as a consequence, should be carefully developed and maintained by the Operating Organization.

Operating Organizations dealing with the commissioning of a NPP for the first time are advised to collect previous experiences from other operators or to seek support from experienced consultants well in advance of the start of commissioning activities. Information exchange visits and, if possible, participation at other sites during their commissioning phase are powerful means of gaining experience, even if the plant concerned is not a reference plant. Institutions like WANO (the World Association of Nuclear Operators) and IAEA and the plant supplier may help to establish the necessary contacts and agreements, and can also provide further sources of information.

22.15 References

HSE (2008), *Applying for a Nuclear Site Licence for New Nuclear Power Stations: A Step-by-Step Guide*, Health and Safety Executive, London (available at www.hse.gov.uk/newreactors/guidance.htm).

IAEA (1996), *Nuclear Power Plant Personnel and its Evaluation: A Guidebook*, Technical Report Series TRS-380, IAEA, Vienna (available at www.IAEA.org/publications).

IAEA (2000), *Safety of Nuclear Power Plants: Operation*, Safety Standards Series NS-R-2, IAEA, Vienna (available at www.IAEA.org/publications).

IAEA (2002), *Maintenance, Surveillance and In-service Inspection in Nuclear Power Plants*, Safety Standards Series, NS-G-2.6, IAEA, Vienna (available at www.IAEA.org/publications).

IAEA (2003), *Commissioning of Nuclear Power Plants*, Safety Standards Series NS-G-2.9, IAEA, Vienna (available at www.IAEA.org/publications).

IAEA (2008), *Best Practices in the Organization, Management and Conduct of an Effective Investigation of Events at Nuclear Power Plants*, TECDOC-1600, IAEA, Vienna (available at www.IAEA.org/publications).

STUK (2003), *The Commissioning of a Nuclear Power Plant*, YVL 2.5, STUK, Finland (available at www.stuk.fi).

23
Operational safety of nuclear power plants

M. LIPÁR, International Atomic Energy Agency (IAEA), Austria

Abstract: This chapter describes the IAEA safety requirements for operation of nuclear power plants. It deals with the management and organizational structure of the operating organization and the management of operational safety, as well as the related requirements for safety programmes, plant operation, and maintenance, testing, surveillance and inspection. In addition, more detailed expectations for different operational areas are described, such as sound policies, procedures, processes and practices, training, technical support, operating experience, chemistry, radiation protection, fire protection and emergency planning and preparedness, the capability and reliability of the operating personnel, comprehensive instructions and adequate resources. Finally, the IAEA Operational Safety Review Team (OSART) is briefly described.

Key words: operational safety, operating organization, operational limits and conditions, operating procedures, training, maintenance.

23.1 Introduction

Operational safety is one of the most challenging areas in nuclear safety. In addition to having to consider sound engineering and technology principles, it is necessary to take into account the human and organizational factors that can either contribute to, or detract from, safety. There is extensive IAEA documentation on operational safety, including safety fundamentals, safety requirements and supporting safety guides, and guidelines for related safety services, in particular for the Operational Safety Review Team (OSART), which has been one of the most often requested safety services of the Agency over the years.

Published by Woodhead Publishing Limited, 2012

At the time of writing (31 January 2011) there are 442 nuclear power reactors in operation worldwide with a total net installed capacity of 374.973 GW(e) (Fig. 23.1), and 65 nuclear power reactors are under construction (Fig. 23.2).

Section 23.2 of this chapter presents a summary of the safety requirements for operational safety of nuclear power plants. Sections 23.3–23.11 provide a summary of the expectations for performance in several important areas of operational safety. Section 23.12 provides information about the Operational Safety Review Team (OSART), Section 23.13 provides sources of additional information, and finally Section 23.14 contains references.

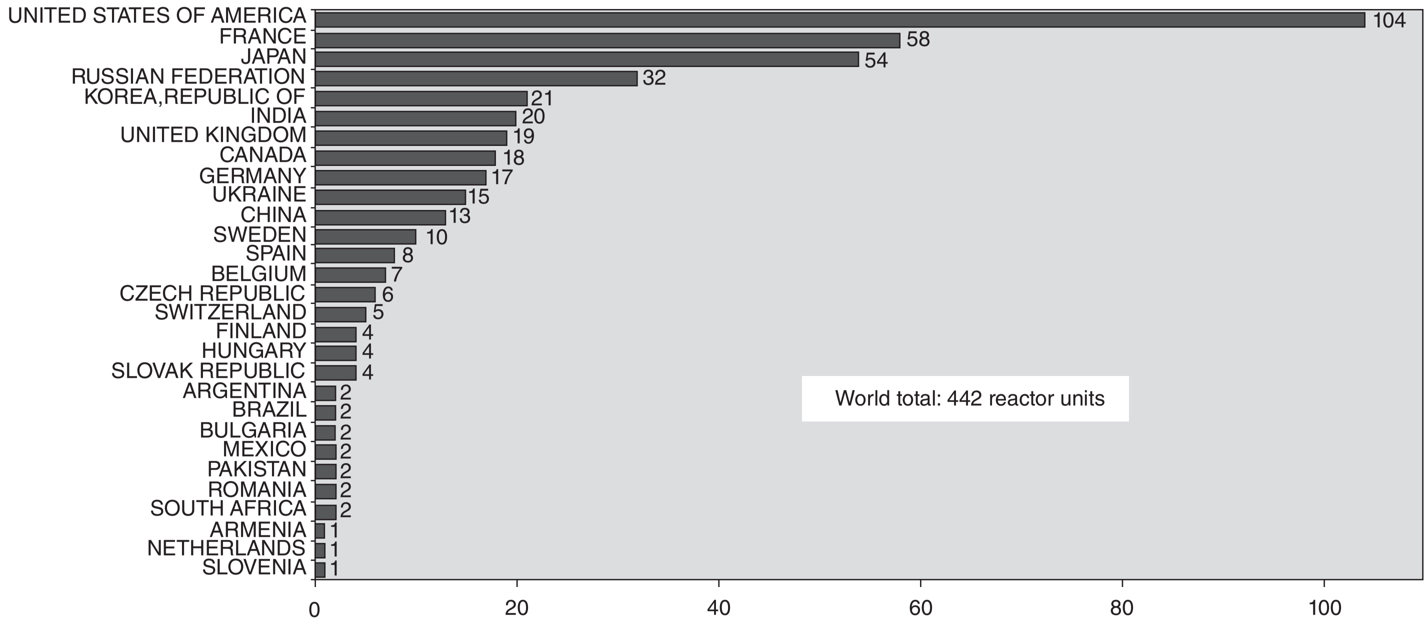
23.2 International Atomic Energy Agency (IAEA) requirements for nuclear power plant (NPP) operation

The International Atomic Energy Agency (IAEA), in its Safety Standards Series publications, has developed requirements for the safe operation of nuclear power plants (IAEA, 2011a). Such requirements cover operation, commissioning and preparation for decommissioning. The last two are considered in other chapters of this book; those for operation follow.

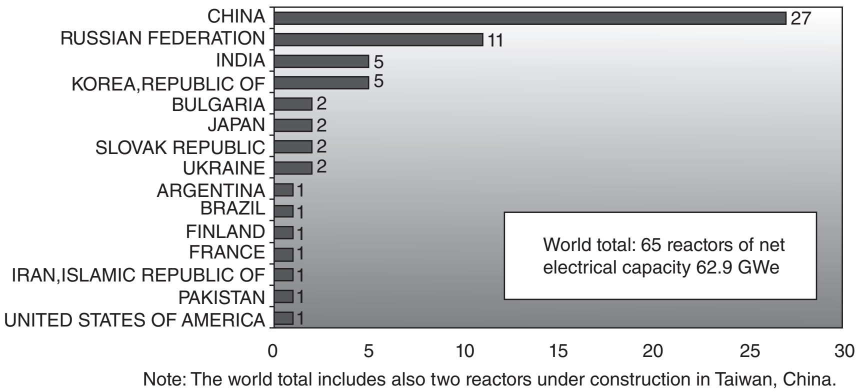
23.2.1 The management and organizational structures of the operating organization

Principle 1 of the IAEA Fundamental Safety Principles (IAEA, 2006a) establishes that the plant licensee is responsible for the safety of the plant and that such responsibility cannot be delegated. Requirement 1 of the operating requirements (IAEA, 2011a) declares that the operating organization shall have the prime responsibility for safety in the operation of a nuclear power plant. The operating organization shall discharge this responsibility in accordance with its management system.

Principle 3 of the IAEA Fundamental Safety Principles (IAEA, 2006a) establishes that effective leadership and management for safety must be established and sustained in organizations concerned with, and facilities and activities that give rise to, radiation risks. Requirement 2 of the operating requirements (IAEA, 2011a) declares that the operating organization shall establish, implement, assess and continually improve an integrated management system. The management system shall integrate all the elements of management so that processes and activities that may affect safety are established and conducted coherently with other requirements, including requirements in respect of leadership, protection of health, human performance, protection of the environment, security and quality.



23.1 Number of reactors in operation worldwide.



23.2 Number of reactors under construction worldwide.

Requirement 3 of the operating requirements (IAEA, 2011a) declares that the structure of the operating organization and the function, roles and responsibilities of its personnel shall be established and documented. Functional responsibilities, lines of authority, and lines of internal and external communication for the safe operation of a plant in all operational states and in accident conditions shall be clearly specified in writing.

In addition, requirement 4 of the operating requirements (IAEA, 2011a) declares that the operating organization shall be staffed with competent managers and sufficient qualified personnel for the safe operation of the plant. The operating organization shall be responsible for ensuring that the necessary knowledge, skills, attitudes and safety expertise are sustained at the plant, and that long-term objectives for human resources policy are developed and met.

23.2.2 Management of operational safety

Requirement 5 of the operating requirements (IAEA, 2011a) declares that the operating organization shall establish and implement operational policies that give safety the highest priority. The safety policy shall stipulate clearly the leadership role of the highest level of management in safety matters and shall give safety the utmost priority, overriding the demands of production or project schedules. This policy shall promote a strong safety culture including a questioning attitude and a commitment to excellent performance in all activities important to safety.

Requirement 6 of the operating requirements (IAEA, 2011a) declares that the operating organization shall ensure that the plant is operated in accordance with the set of operational limits and conditions. They shall reflect the provisions made in the final design as described in the safety analysis report. The plant shall be operated within operational limits and conditions to prevent situations arising that could lead to anticipated operational occurrences or accident conditions.

Requirement 7 of the operating requirements (IAEA, 2011a) declares that the operating organization shall ensure that all activities that may affect safety are performed by suitably qualified and competent persons. The operating organization shall clearly define the requirements for qualification and competence. Suitably qualified personnel shall be selected and shall be given the necessary training and instruction to enable them to perform their duties correctly for different operational states. Certain operating positions may require formal authorization or a licence.

Principle 5 of the IAEA Fundamental Safety Principles (IAEA, 2006a) establishes that protection must be optimized to provide the highest level of safety that can reasonably be achieved. Requirement 8 of the operating requirements (IAEA, 2011a) declares that the operating organization shall ensure that safety-related activities are adequately analysed and controlled to ensure that the risks associated with harmful effects of ionizing radiation are kept as low as reasonably achievable. All routine and non-routine operational activities shall be assessed for the potential risks associated with harmful effects of ionizing radiation.

Requirement 9 of the operating requirements (IAEA, 2011a) declares that the operating organization shall establish a system for continuous monitoring and periodic review of the safety of the plant and of the performance of the operating organization. An adequate audit and review system shall be established to ensure that the safety policy of the operating organization is being implemented effectively and that lessons are being learned from its own experience and from the experience of others to improve safety performance. 'Self-assessment' by the operating organization shall be an integral part of the monitoring and review system.

Requirement 10 of the operating requirements (IAEA, 2011a) declares that the operating organization shall establish and implement a system for plant configuration management to ensure consistency between design requirements, physical configuration and plant documentation. Controls on plant modifications shall ensure that changes to the plant and its safety-related systems are properly identified, specified, screened, designed, evaluated, authorized, implemented and recorded.

Requirement 11 of the operating requirements (IAEA, 2011a) declares that the operating organization shall establish and implement a programme to manage modifications. Modification programmes shall cover structure, systems, and components, operational limits and conditions, procedures, documents and the structure of the operating organization. Modifications shall be characterized on the basis of their safety significance. Temporary modifications shall be limited in time and number to minimize the cumulative safety significance.

Requirement 12 of the operating requirements (IAEA, 2011a) declares that systematic safety assessments of the plant in accordance with the regulatory requirements shall be performed by the operating organization throughout the plant's operational lifetime, with due account taken of operating experience and significant new safety-related information from all relevant sources. The scope of the safety review shall include all safety-related aspects of an operating plant.

Requirement 13 of the operating requirements (IAEA, 2011a) declares that the operating organization shall ensure that a systematic assessment is carried out to provide reliable confirmation that safety-related items are capable of the required performance for all operational states and for accident conditions. Appropriate concepts and the scope and process of equipment qualification shall be established, and effective and practicable methods shall be used to upgrade and preserve equipment qualification.

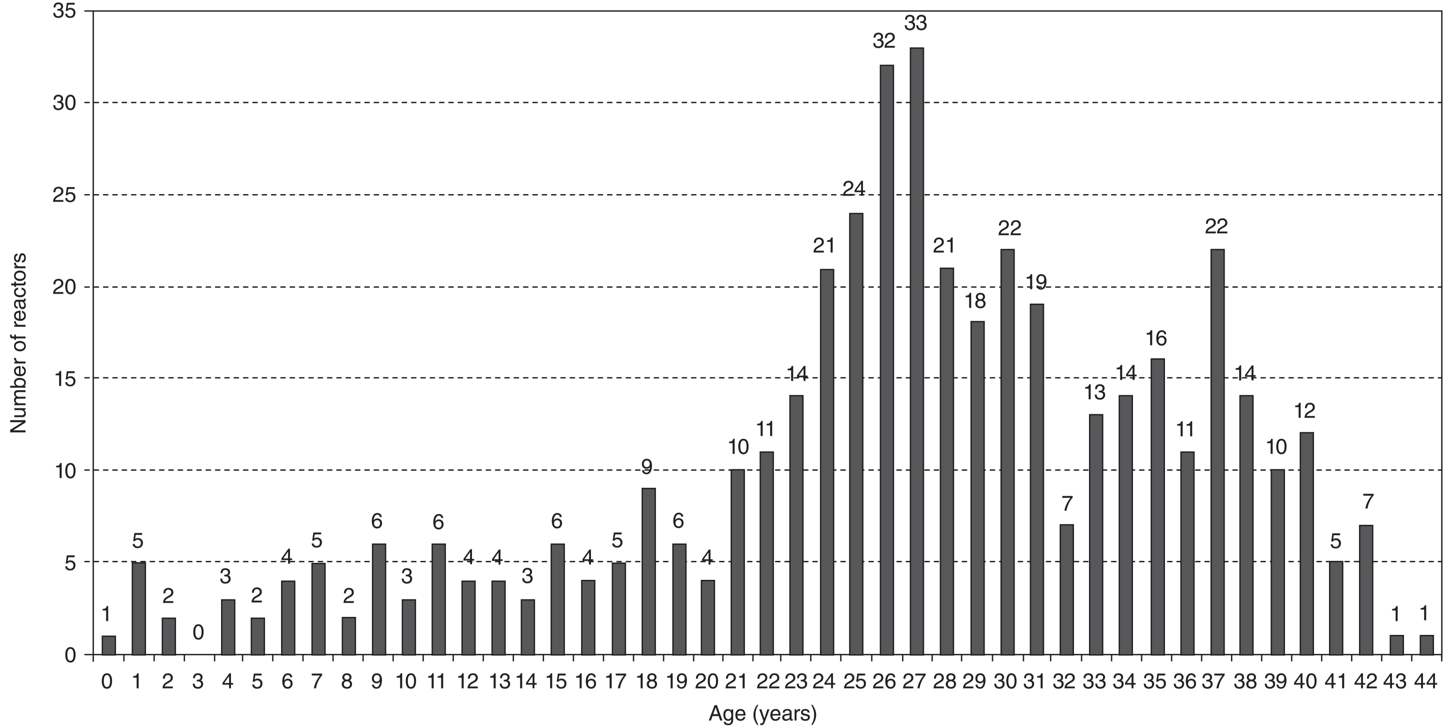
Requirement 14 of the operating requirements (IAEA, 2011a) declares that the operating organization shall ensure that an effective ageing management programme is implemented to ensure that required safety functions of systems, structures and components are fulfilled over the entire operating lifetime of the plant. The ageing management programme shall determine the consequences of ageing and the activities necessary to maintain the operability and reliability of structures, systems and components.

Requirement 15 of the operating requirements (IAEA, 2011a) declares that the operating organization shall establish and maintain a system for the control of records and reports. The operating organization shall identify the types of records and reports, as specified by the regulatory body, that are relevant for the safe operation of the plant.

Requirement 16 of the operating requirements (IAEA, 2011a) declares that, where applicable, the operating organization shall establish and implement a comprehensive programme for ensuring the long-term safe operation of the plant beyond a time-frame established in the licence conditions, design limits, safety standards and/or regulations. The justification for long-term operation shall be prepared on the basis of the results of a safety assessment, with due consideration of the ageing of structures, systems and components, and should utilize the results of periodic safety review. Figure 23.3 shows the age distribution of existing NPPs.

23.2.3 Operational safety programmes

Requirement 17 of the operating requirements (IAEA, 2011a) declares that the operating organization shall ensure that the implementation of safety requirements and security requirements satisfies both safety objectives and security objectives. Safety and security measures shall be designed and implemented in such a way that they do not compromise each other.



Note: Age of a reactor is determined by the year of its first grid conection.

23.3 Number of operating reactors by age.

Principle 9 of the IAEA Fundamental Safety Principles (IAEA, 2006a) establishes that arrangements must be made for emergency preparedness and response for nuclear or radiation incidents. Requirement 18 of the operating requirements (IAEA, 2011a) declares that the operating organization shall prepare an emergency plan for preparedness for and response to a nuclear or radiological emergency. Emergency preparedness arrangements shall cover the capability of maintaining protection and safety in the event of accident conditions, mitigating the consequences of accidents if they occur, protection of site personnel and the public, and protection of the environment, as well as coordinating response organizations, as appropriate, and communicating with the public in a timely manner.

Principle 8 of the IAEA Fundamental Safety Principles (IAEA, 2006a) establishes that all practical efforts must be made to prevent and mitigate nuclear or radiation accidents. Requirement 19 of the operating requirements (IAEA, 2011a) declares that the operating organization shall establish an accident management programme for the management of beyond-design-basis accidents. The programme shall cover the preparatory measures and guidelines that are necessary for dealing with beyond-design accidents. Arrangements for accident management shall provide the operating staff with appropriate systems and technical support.

Principle 7 of the IAEA Fundamental Safety Principles (IAEA, 2006a) establishes that people and the environment, present and future, must be protected against radiation risks. Requirement 20 of the operating requirements (IAEA, 2011a) declares that the operating organization shall establish and implement a radiation protection programme. The programme shall be in compliance with the requirements of the International Basic Safety Standards (IAEA, 1996).

Requirement 21 of the operating requirements (IAEA, 2011a) declares that the operating organization shall establish and implement a programme for the management of radioactive waste. Adequate operating practices shall be implemented to ensure that the generation of radioactive waste is kept to the minimum practicable in terms of both activity and volume.

Requirement 22 of the operating requirements (IAEA, 2011a) declares that the operating organization shall make arrangements for ensuring fire safety. The arrangements shall cover adequate management for fire safety, preventing fires from starting, detecting and extinguishing quickly any fires that do start, preventing the spread of those fires that have not been extinguished, and providing protection from fire from structures, systems and components that are necessary to shut down the plant safely.

Requirement 23 of the operating requirements (IAEA, 2011a) declares that the operating organization shall establish and implement a programme to ensure that safety-related risks associated with non-radiation-related hazards to personnel involved in activities at the plant are kept as low as reasonably achievable. All personnel, suppliers, contractors and visitors (where appropriate) shall be trained and shall have the necessary knowledge of the non-radiation-related safety programme (industrial safety).

Requirement 24 of the operating requirements (IAEA, 2011a) declares that the operating organization shall establish an operational experience programme to learn from events at the plant and events in the nuclear industry and other industries worldwide. The programme shall cover reporting, collecting, screening, analysing, trending, documenting and communicating operational experience at the plant in a systematic way.

23.2.4 Plant operations

Requirement 26 of the operating requirements (IAEA, 2011a) declares that the operating procedures shall be developed that apply comprehensively (for the reactor and its associated facilities) for normal operation, anticipated operational occurrences and accident conditions, in accordance with the policy of the operating organization and the requirements of the regulatory body. The procedures and reference material shall be clearly identified and shall be readily accessible in the control room and in other operating locations if necessary.

Requirement 27 of the operating requirements (IAEA, 2011a) declares that the operating organization shall ensure that the operation control rooms and control equipment are maintained in a suitable condition. The habitability and good condition of control rooms shall be maintained.

Requirement 28 of the operating requirements (IAEA, 2011a) declares that the operating organization shall develop and implement programmes to maintain a high standard of material conditions, housekeeping and cleanliness in all working areas.

Requirement 29 of the operating requirements (IAEA, 2011a) declares that the operating organization shall establish and implement a chemistry programme to provide the necessary support for chemistry and radiochemistry. The chemistry programme shall provide the necessary information and assistance for chemistry and radiochemistry for ensuring safe operation, long-term integrity of structures, systems and components, and minimization of radiation levels.

Requirement 30 of the operating requirements (IAEA, 2011a) declares that the operating organization shall be responsible and shall make arrangements for all activities associated with core management and with on-site fuel handling. Provisions shall be made to ensure that only fuel that has been appropriately manufactured is loaded into the core. The fuel design criteria and fuel enrichment shall be in accordance with design specifications and shall be approved by the regulatory body as required.

23.2.5 Maintenance, testing, surveillance and inspections

Requirement 31 of the operating requirements (IAEA, 2011a) declares that the operating organization shall ensure that effective programmes for maintenance, testing, surveillance and inspection are established and implemented. Maintenance, testing, surveillance and inspection programmes shall be established that include predictive, preventive and corrective maintenance activities. These maintenance activities shall be conducted to maintain availability during the service life of structures, systems and components by controlling degradation or preventing failures.

Requirement 32 of the operating requirements (IAEA, 2011a) declares that the operating organization shall establish and implement arrangements to ensure the effective performance, planning and control of work activities during outages. Outage planning shall be a continuing, improving process involving past, present, next scheduled and future outages.

23.3 Management, organization and administration of nuclear power plants (NPPs)

The organizational structure of a nuclear power plant (NPP) must support safe, reliable and effective performance and control of all power plant activities. The organization of the nuclear power plant provides the administrative and functional structure that determines where people are assigned, what they are to do, and how they are expected to accomplish their tasks. Policies, directives, procedures, goals and objectives and performance standards provide administrative controls and management direction to implement the organizational structure, to conduct all power plant activities and ensure safe operation of the power plant. The organizational structure establishes formal relationships and lines of communication. Responsibilities and authorities for accomplishing assigned tasks should be clearly defined and communicated within the established organizational structure.

Management monitoring and assessment activities are integral parts of the administrative system to identify areas where performance is achieving the high standards expected by management as well as where performance is deviating from management expectations.

In addition, a sound safety management system should be established at the power plant as an integral part of the overall management system. The safety management system should comprise those arrangements made by the operating organization that are needed to promote a strong safety culture and achieve and maintain good safety performance.

For this purpose the management, organization and administration includes NPP management practices as well as the quality assurance programme, the industrial safety programme, and document and records management that are also important elements of NPP management and contribute to the safe operation of the NPP.

More information is in IAEA (2011a) and IAEA (2001b).

23.3.1 Organization and administration

The operating organization should establish for the plants under its control an organizational plan that indicates the general policies, lines of responsibility and authority, lines of communication, duties and number of staff and their required qualifications needed to run the plants. When new construction, retirement or other developments indicate that some critical plant personnel may leave the workforce, management should have plans for filling the openings with competent people.

The plant's documented organizational structure shall indicate the staffing arrangements within the categories of direct line operating personnel and supporting personnel. Functional responsibilities, levels of delegated authority and lines of internal and external communication for safe operation of the plants in all operational states, for mitigating the consequences of accident conditions and for ensuring an appropriate response in emergencies, shall be clearly defined in writing. The extent to which the support functions are self-sufficient or dependent upon services from outside the plant organization shall be shown by means of functional organizational charts which include personnel resource allocations and specify the duties and responsibilities of key personnel. Likewise, the transfer of responsibility across interfaces should be clearly defined and understood.

Adequate financial and manpower resources and facilities should be made available to managers for safe and efficient operation of the plant. Adequate provisions of qualified spares, materials and equipment should be consistent with the need for timely execution of safety-related activities. The management system should be supported by a well-established human resources management programme that includes high standards for recruitment and selection of personnel, a well-established performance appraisal system, and a promotion and succession-planning system that takes into account attitudes towards safety. A fitness-for-duty policy should be established that ensures individuals are physically and mentally fit to perform their job in a safe manner.

Suitably qualified and experienced persons shall perform all activities that may affect safety. The nuclear power plant shall be staffed with competent managers and a sufficient number of qualified personnel having a proper awareness of the technical and administrative requirements for safety and motivated to be safety conscious. Attitudes towards safety shall be a criterion for the hiring or promoting of managers. Staff performance appraisals shall also include the attitude towards safety.

Supporting activities provided by contractors should adhere to the same standards as plant quality and safety policies. The plant requirements relating to quality and competency of the contractor staff and work product should be at the same standard as the activities carried out by the plant staff. Contractors' staff shall be properly controlled and supervised by the plant staff.

To enable the regulatory body to perform its functions, the operating organization shall render all necessary assistance and shall grant access to the plant and documentation. Mutual understanding and respect between the regulatory body and the operating organization, and a frank, open and yet formal relationship, shall be fostered.

Recently in many countries the nuclear industry has been going through a period of significant changes. These changes arise from the political and business environment in which the industry must operate, and from within the industry itself as it strives to become more competitive. Changes to staffing levels, ways of working or organizational structure should be subject to analysis and independent review when proposed. These changes must be carefully considered with respect to potential impacts on nuclear safety. Changes should be monitored during and after implementation to ensure that they are not detrimental to safety. The need for change should be communicated to the staff and ownership of the need for change established with those involved.

More information is in IAEA (2001b), IAEA (2002d), IAEA (2006c), IAEA (2006d), IAEA (2010a) and IAEA (2011a).

23.3.2 Management activities

Management should establish and clearly communicate high standards of performance to promote excellence in the conduct of all power plant activities. Management policies and directives covering conduct of activities should reflect desired high standards. In particular, there should be a clear statement of quality and safety policy according to senior management's commitment. Goals and objectives that promote excellence in plant operation and focus on areas needing improvement should be in place. Good communication of management expectations should be established within the plant and also with outside organizations.

Managers should actively promote and frequently reinforce corporate policies, safety goals and objectives. Plant management should develop goals and objectives that support and complement established corporate goals. Suitable goals and objectives should be established at departmental level to support the goals of the plant management. Where it is reasonable, the goals and objectives of all management levels should be measurable and stated in terms that allow measurement of progress and clear determination of achievement.

Supervisors and managers should fully understand their role and responsibilities and the reasons for required policies. They should display those values and behaviours required to demonstrate that safety is their top priority. A mechanism should exist for plant staff to report safety concerns to management. There should also be a mechanism for staff to report safety concerns to an independent body (e.g. regulator) if they are not satisfied with management response. Senior-level managers should be accessible and respond to suggestions from personnel. Managers should routinely be in the field to assess and discuss the conduct of work and compliance with management objectives.

Administrative procedures, rules and instructions, covering all aspects of plant operation and applicable to all personnel on site, should ensure safe and effective methods of working and uniformity of performance.

Priorities of management efforts and resource allocation should reflect the safety significance of the issues dealt with, and the risks associated with them. Probabilistic safety assessments (PSAs) have been performed by many nuclear power plant organizations to identify potential plant vulnerabilities and understand the relative risk contribution of particular design and operational features. As a result of the availability of PSA studies, there is a desire to use them to enhance plant safety and to operate the nuclear stations more efficiently. PSA has proved to be an effective tool for this purpose as it assists plant management to target resources where the largest benefit to plant safety can be obtained. The current state-of-the-art in PSA is considered to be sufficiently well developed that the insights from such studies can be used sensibly in the plant safety decision-making process and risk management. However, any PSA that is to be used for such a purpose must have a credible and defensible basis.

More information is in IAEA (2001b), IAEA (2002d), IAEA (2006c), IAEA (2006d) and IAEA (2011a).

23.3.3 Safety management

This section on the management of safety should not be taken to suggest that safety is managed separately from other management activities. Neither should it be seen as an optional extra. The organization's safety management system is generally considered to be an integral part of its overall management system.

A safety management system should be applied, integrating management of safety, health, environmental quality and economic matters in a coherent manner.

A policy on safety shall be developed by the operating organization and applied by all site personnel. This policy shall give safety the utmost priority at the plant, overriding if necessary the demands of production and project schedules. The safety policy should demonstrate the organization's commitment to high safety performance and be supported by reference to safety standards, the development of targets and provision of the resources necessary to achieve these targets. The policy should be provided to all staff members for their guidance and clearly understood by all of them and declared to the public as one of the objectives of the operating organization. The operating organization should ensure that adequate resources are available to implement the safety policy.

All functions in the operating organization should encourage and support sound safety management practices at the highest levels of corporate and plant management. Managers, at various organizational levels, should demonstrate their commitment to safety as a top priority.

The risks associated with any operating activity at the plant should be systematically evaluated and measures taken to eliminate or mitigate the identified risks.

The operating organization should demonstrate a commitment to achieving improvements in safety wherever it is reasonably practicable to do so as part of a continuing commitment to the achievement of excellence. The organization's improvement strategy for achieving higher safety performance and for more efficient ways to achieve existing standards should be based on a well-defined programme with clear objectives and targets against which to monitor progress.

The operating organization should comprehensively monitor plant operation to ensure its licensee accountability and to evaluate performance against the goals and objectives established for safe operation of the plant. Senior plant management should routinely monitor performance against these goals and objectives, and hold responsible staff accountable for their achievement.

Performance indicators should be established to measure the progress in achieving the goals and objectives. They should be regularly assessed against defined goals and objectives, and the results should be communicated to staff and used to derive corrective actions.

More information is in IAEA (1991), IAEA (2000a), IAEA (2001b), IAEA (2002d), IAEA (2002f), IAEA (2006c), IAEA (2006d), IAEA (2009d) and IAEA (2011a).

23.3.4 Quality assurance programme

The operating organization should develop, implement and maintain a quality policy and a quality assurance (QA) programme. The QA programme should serve as a management tool in verifying or confirming, through meaningful monitoring, that the requirements established within the organization are being achieved. This programme should include details of how work is to be managed, performed and assessed. It includes the organizational structure, functional responsibilities, level of authority and interfaces for those managing, performing and assessing the adequacy of the work. The QA programme should address management measures, including planning, scheduling and resource considerations.

Management in the entire and constituent areas of work should provide and demonstrate support for the effective implementation of the QA programme consistent with specified time schedules for accomplishing project activities. The operating organization is responsible for the establishment and implementation of the overall QA programme. If it delegates the work of establishing and implementing all or part of the overall programme, it retains responsibility for the effectiveness of the programme in all circumstances.

Quality assurance requirements should be applied to activities such as operations, maintenance and procurement of replacement items, tests or experiments, changes of configuration and plant modification, which may be undertaken by other units of the operating organization or by external agencies. It should remain the responsibility of plant management to ensure that arrangements are in place to control all activities affecting quality.

Safety issues should be the fundamental consideration in the identification of items, services and processes to which the QA programme applies. A graded approach based on the relative importance to safety of items, services and processes should be used. It should reflect a planned and recognized difference in the applications of specific quality assurance requirements.

Independent assessments should be conducted on behalf of management to measure the effectiveness of management processes and the adequacy of work performance, to monitor item and service quality and to promote improvement.

More information is in IAEA (2006c), IAEA (2006d), IAEA (2009d) and IAEA (2011a).

23.3.5 Industrial safety programme

The operating organization should have a general policy to ensure the industrial health and safety of personnel on site is satisfactory. All elements of this policy should be documented in a plant safety manual, while details are included in implementing procedures.

The industrial safety programme should be known, understood and adhered to by all personnel on site. Senior management should be committed to industrial safety; line supervisors should have the authority and responsibility to ensure good industrial safety performance. A suitable organization should be in place that supports the programme and a process should be implemented that routinely reviews the status of industrial safety practices. A risk analysis should be performed prior to any activity.

More information is in IAEA (2006c), IAEA (2006d), IAEA (2009d) and IAEA (2011a).

23.3.6 Documentation and records management

A documentation and records management system should be established to ensure the appropriate keeping of all documents relevant to the safe and reliable operation of the plant, including design documents, commissioning documents, and documents related to the operational history of the plant, as well as general and specific procedures. Control of documentation should be done in a consistent, compatible manner throughout the plant and the operating organization. This includes preparation, change, review, approval, release and distribution of documentation. Lists and procedures for these functions should be prepared and controlled.

The records system should ensure that records are specified, prepared, authenticated and maintained, as required by applicable administrative procedures in accordance with the QA requirements. Information sources should be integrated, when appropriate, to improve the accuracy, timeliness and availability of the information.

A suitable records storage system should be in place to ensure safe conservation and easy accessibility of all documents and records necessary to operate the plant.

More information is in IAEA (2001b), IAEA (2006c), IAEA (2006d) and IAEA (2011a).

23.4 Training and qualification

To achieve and maintain high safety standards, nuclear power plants are required to be staffed by an adequate number of highly qualified and experienced personnel. To establish and maintain a high level of personnel competence, appropriate training and qualification programmes should be established at the plant and kept under constant review, to ensure their relevance to staff needs. It is the responsibility of the operating organization to ensure that all plant personnel receive appropriate training and that only personnel with suitable qualifications are assigned job functions at the nuclear plant. During employment, qualifications are maintained by participation in continuing training programmes that are directed towards maintaining and upgrading the knowledge and skills of the personnel.

More information is in IAEA (2002d) and IAEA (2011a).

23.4.1 Training policy and organization

The operating organization should formulate an overall training policy. The training policy should be known, understood and supported by all persons concerned. A training plan should be prepared on the basis of the long-term needs and goals of the plant. A systematic approach to training should be used for the training of plant personnel. A system should be in place to identify the training needs of all staff following their recruitment. These training needs should be reviewed and revised to take account of organizational changes and changes in plant and processes. Appropriate mechanisms should ensure that a 'corporate memory' of safety-related events is retained.

The plant manager should be responsible for the qualification of plant staff and should support the training organization with necessary resources including staffing and facilities. He should ensure that cost reduction programmes do not lead to undue limitation of resources being made available for training and retraining staff. Succession planning should be an established practice in the training organization. The training organization should be responsible for assisting the plant manager in establishing, verifying and maintaining the competence of plant staff. The training organization should be well defined, including its interfaces with other plant groups. Line managers and supervisors should be accountable for the qualification of their personnel and involved in defining their training needs and ensuring that the training provided reflects operating experiences. Managers and supervisors should ensure that production requirements do not interfere with the conduct of training programmes.

The operating organization should ensure that the qualifications and training of external personnel performing safety-related duties are adequate for the functions to be performed.

Qualifications of each individual should be assessed against established training objectives and performance criteria during and after the training and before assignment to a new job and periodically thereafter. Individual training records should be maintained. Persons performing certain functions important to safety should be required to hold a formal authorization.

The Institute of Nuclear Power Operation (INPO) provides the programme on accreditation of the training programmes in each of the US nuclear power plants. This highly involved accreditation programme is now being considered and put in practice in some countries, notably the UK and more recently in Spain.

More information is in IAEA (2002d) and IAEA (2011a).

23.4.2 Quality of the training programmes

Performance-based programmes for initial and continuing training shall be developed and put in place for each major group of personnel. The content of each programme should be based on a systematic approach, such as job and task analysis, ensuring the necessary knowledge and skills are incorporated. Training programmes should be in place to address safety culture. Such programmes should stress that individuals understand the significance of their duties and the consequences of mistakes arising from misconceptions or lack of diligence. Training programmes shall promote attitudes that help to ensure that issues of safety receive the attention they warrant. Training programmes for most NPP positions should include periods of formal training in the classroom intermixed with intervals of simulator, or laboratory, or workshop, training and should include practical training in the plant. This training should be conducted and evaluated in the work environment by qualified, designated individuals.

The adequacy of all training programmes should be periodically reviewed and assessed by both plant management and the training staff. This should include evaluation of training graduate competence in the workplace and adjustment of training programmes as necessary. The programme should be designed to allow for updating when changes in the tasks, plant systems or procedures are made. In addition, a system shall be in place for timely modification and updating of the training facilities and materials to ensure that they accurately reflect plant conditions.

More information is in IAEA (1991), IAEA (2002d), IAEA (2002f) and IAEA (2011a).

23.4.3 Training programmes for control room operators and shift supervisors

The training and qualification programme for control room operator (CRO) and shift supervisor (SS) should develop and improve the competence to operate the controls of a nuclear power plant and direct those who manipulate the controls in the control room and in the plant.

Their training programme should develop and maintain adequate knowledge and skills to ensure that they are able to:

- Monitor and control the plant system status in accordance with relevant rules, operating instructions, technical specifications and administrative procedures

- Conduct all operations in a safe and reliable manner, without causing excessive thermal or mechanical load to the plant equipment
- Take correct actions in response to various abnormal conditions, and bring the plant to a safe condition, including shutdown, whenever needed.

The training programmes should also include broad knowledge of the fundamentals to provide a basis for understanding the operation of systems and integrated plant operations and to diagnose system/component problems.

More information is in IAEA (2002d) and IAEA (2011a).

23.4.4 Training programmes for field operators

The field operator training and qualification programme should develop, maintain and improve the knowledge and skills necessary to operate equipment outside the control room in accordance with relevant instructions and procedures, as directed by the control room staff. This training programme should develop and maintain basic knowledge and skills in similar areas as the programme for control room operators but it should emphasize practical work-specific topics. Well-trained field operators should be able to:

- Monitor the equipment performance and status in the field and recognize any deviations from the normal conditions
- Conduct all field operations in a safe and reliable manner, without causing unacceptable risks to plant
- Detect and properly respond to plant conditions with the goal of preventing or, at minimum, of mitigating unanticipated plant transients.

More information is in IAEA (2002d) and IAEA (2011a).

23.4.5 Training programmes for maintenance personnel

The training and qualification programme for maintenance personnel should develop and maintain or improve the knowledge and skills necessary for carrying out preventive and predictive maintenance, repairs and plant modifications. Training programmes for maintenance personnel should include plant layout and the general features and purposes of plant systems, quality assurance and quality control, maintenance procedures and practices, including surveillance and inspections, and special maintenance skills. An appropriate emphasis on the safety culture should be included in all aspects of training for maintenance personnel. Training programmes for maintenance personnel should emphasize the potential safety consequences of technical or procedural errors. Experience of faults and hazards caused by errors in maintenance procedures and practices at the NPP or at other plants and in other industries should be reviewed and incorporated into training programmes as appropriate.

Special training provided to individuals should develop their craft skills and ensure qualification on equipment to which they are assigned to work.

More information is in IAEA (2002d) and IAEA (2011a).

23.4.6 Training programmes for technical plant support personnel

The training and qualification programmes for technical support personnel based on the specific needs of the power plant should be established to develop and maintain the knowledge and skills of technical personnel to support safe and reliable plant operation. Consideration also should be given to the training needs of contracted personnel to ensure that the requirements of the operating organization are met. Technical support personnel should acquire knowledge of plant systems and understanding of operational methods and environment, so that they can effectively guide and interact with operating and maintenance personnel. These personnel should have knowledge of the operational features of the plant and preferably possess 'hands on' experience. In addition to technical training, appropriate training in other areas, such as supervisory and communication skills, should be provided. Dependent on the specific technical support groups, the appropriate training programmes should cover such subject areas as reactor physics and core management, chemistry, radiation protection, surveillance and testing, planning, performance and plant engineering, safety analyses and reviews, emergency preparedness, records administration and documentation, and quality assurance.

More information is in IAEA (2002d) and IAEA (2011a).

23.4.7 Training programmes for management and supervisory personnel

The plant should have a management development programme to ensure that an adequate number of experienced and qualified staff is available to fill any manager or supervisor position, in the event that a position is unexpectedly vacated. Training programmes for management and supervisory personnel should emphasize the concept and practices of safety culture. These programmes should emphasize the special problems of managing an NPP, with the exceptional demand for safety and the need for familiarity with emergency procedures. They should give a thorough understanding of relevant standards, rules and regulations. They should also give a good overall knowledge of the plant and its systems. The managers and supervisors with responsible positions in the emergency preparedness organization should be specially trained for their emergency duties. Special attention should be given to gaining from the benefits of operational experience feedback and root-cause analysis for events that are generic or occur frequently at the plant. Training programmes for managers and supervisors, and their potential successors, should also include courses and seminars on management and supervisory skills, coaching and mentoring, decision making, self-assessment techniques, root-cause analysis, team training, and communications. The managers and supervisors should also attend continuing training in their areas of responsibility, in order to maintain current technical knowledge and to be able to supervise training of their staff.

More information is in IAEA (2002d) and IAEA (2011a).

23.4.8 Training programmes for training group personnel

All training department staff, simulator and technical support engineers, technicians and instructors should be given training commensurate with their duties and responsibilities. Training instructors shall be technically competent in their assigned areas of responsibility and have credibility with the trainees and other plant personnel. They should understand all aspects of the content being taught and the relationship of that content to overall plant operation. In addition, the instructors should be familiar with the basics of adult learning and of a systematic approach to training and have adequate instructional and assessment skills. Instructors should also be given the time necessary to maintain their technical and instructional competence, by secondment or attachment to operating plant on a regular basis, and by continuing training. Personnel in the on-site training department should also be properly trained in matters concerning the policies of the operating organization, in particular safety management and safety culture, the regulatory requirements and quality assurance.

More information is in IAEA (2002d) and IAEA (2011a).

23.4.9 General employee training

All new employees starting work at nuclear power plants should be introduced to the organization and their work environment in a systematic and consistent manner. General employee training (GET) programmes should give new employees a basic understanding of their responsibilities and safe work practices, the importance of quality programmes and following procedures and the practical abilities to protect themselves from hazards associated with their work. Hands-on training in radiation protection actions, which are common to all plant personnel, should be provided to all who work in radiological controlled areas. The depth of the knowledge to be provided on each topic should be commensurate with the duty and position of the person. The basic principles of safety culture should be taught to all employees. Refresher training on GET topics should also be periodically provided.

The operating organization should ensure that contractor personnel involved in safety-related activities are competent, qualified and medically fit to perform their assigned tasks.

All suppliers and contractors involved in design, engineering, manufacturing, construction, operation, maintenance or other safety-related activities should be aware of the applicable standards while working at a nuclear power plant or for an operating organization. Suppliers and contractors should understand the safety culture demonstrated by the plant personnel.

23.5 Operations

Operations involve activities that supervise the operating group which controls safe plant operation. Their main function is to run the plant safely and efficiently while adhering to approved procedures, operational limits and conditions (OLCs) and other regulatory requirements.

The operating group has a direct impact on the reactor operations and its associated components and systems through conduct of operations. While the structure of the group varies according to the specific plant or utility, the group is normally composed of shift crews and supporting staff during office hours and is usually managed by a head of operations. The shift supervisor manages plant operations on each shift. During off-hours the shift supervisor maintains the authority of the plant manager. In addition to this, for the purpose of defining review responsibilities in these guidelines, operations covers operation facilities, operator aids, work authorization, fire protection and accident conditions.

More information is in IAEA (2000c), IAEA (2008a) and IAEA (2011a).

23.5.1 Organization and functions

The organization and functions of the direct operating group should ensure that the nuclear power plant is operated safely and conservatively under all operational states and accident conditions. This should include preparation to deal with severe accident conditions.

The organization, qualifications and number of operations personnel should be sufficient for the safe and reliable operation of the plant at power and during shutdowns and outage periods. Succession planning should be an established practice in the operating group.

The responsibilities and authorities of the direct operating group should be clearly defined and understood by all affected personnel.

The operations goals and objectives should be written and defined within the framework of plant policies and be well understood by the operating personnel. In those it should be clear that nuclear safety has an overriding priority. Performance indicators should be established that encourage these expectations and are reported in periodic assessments.

Plant management should be clearly committed to nuclear safety in plant operations. The frequent presence of management in the field will demonstrate this commitment. Leadership and coaching should contribute to the improvement of safety performance.

More information is in IAEA (2001b), IAEA (2009c) and IAEA (2011a).

23.5.2 Operations facilities and operator aids

The facilities and equipment used by the operating staff should be well maintained and adequate to support safe and reliable operation of the plant under all operating conditions.

There should be a programme to control operator aids at the plant. This programme should ensure reliable communications, well-identified and labelled equipment, clearly identified defective or unavailable equipment, good environmental conditions at the plant, clear and accessible information systems and adequate and well-maintained supporting equipment.

More information is in IAEA (2008a) and IAEA (2011a).

23.5.3 Operating rules and procedures

Operating personnel should operate the plant safely and reliably while keeping the plant's operation within the OLCs, in accordance with the policy of the operating organization and the requirements of the regulatory body. Comprehensive legible operating procedures should be provided for the operators.

Procedures shall be developed for normal operation to ensure that the plant is operated within the OLCs. Either event-based or symptom-based procedures shall be developed for anticipated operational occurrences and design-basis accidents. Emergency operating procedures or guidance for managing severe accidents (beyond the design basis) shall be developed.

Guidance provided in the procedures should be clear, concise, verified for its accuracy and validity and adequate to enable trained operators to perform their activities.

All procedures should be properly approved by plant management, controlled by established procedures, and implemented in a timely manner.

Operators should be appropriately trained on procedures, including changes to existing procedures or new procedures.

Changes to plant procedures should only be performed following an approved procedure that designates the appropriate authorities that must approve the change to the procedure.

An appropriate surveillance programme should be established and implemented to ensure compliance with the OLCs, and to ensure that its results are evaluated and retained.

At a multiple unit site, documents and procedures should be located at each unit. Procedures should be written to specifically address which unit or component will be manipulated.

More information is in IAEA (2000c), IAEA (2002b), IAEA (2009c) and IAEA (2011a).

23.5.4 Conduct of operations

Operations personnel should be cognizant of and have control over the status of plant systems and equipment in all modes of operation. The shift supervisor should be informed of all the plant activities affecting the status of systems and components. All activities such as performance and results of surveillance tests and maintenance works should be routed via him or his delegate for final approval. Similarly, the operators should be kept informed of plant status. A policy should be in place that gives direction to the operators on procedure rules and requirements of how a procedure should be used. This policy should include directions for when procedures are to be used as general guidance, are to be followed step-by-step, or need to be signed off for each step. Close adherence to written procedures should be observed in order to ensure correct operation of equipment. The policy should also include directions when a procedure must be physically at the job site, and what actions are to be taken when procedures conflict or are inadequate. Deviation from these procedures should require approval at a level appropriate to its safety significance. Procedure users should be encouraged to provide feedback to procedure writers on inaccuracies, difficulties in use and suggestions for improvement.

The operating department's policies and procedures should reflect an attitude of safe conservative operations. Managers and supervisors should demonstrate and require a conservative approach towards activities affecting the reactor core and safety systems.

Control room activities should be conducted in a businesslike and professional manner. An atmosphere conducive to safe and reliable operation should be maintained. Operators should be alert and attentive to control board indications and alarms. Administrative duties assigned to control room operators should not interfere with their ability to monitor plant parameters and conduct other operational activities. Control room access should be limited to persons on official business only.

The shift crews should routinely monitor the condition of systems and components and make the appropriate records. The important information on the plant status and the relevant operating occurrences should be adequately logged. The operational personnel should conduct regular plant tours to ensure that the status of equipment is evaluated appropriately and abnormal conditions identified. Operational personnel should take appropriate actions to correct or report deficiencies noted during tours.

The shift turnovers should be carried out in accordance with the formal procedure. The procedures should identify the persons involved, their responsibilities, the locations and the conduct of shift turnovers, and methods of reporting plant status, including provisions for special circumstances such as abnormal plant status and staff unavailability.

Effective reviews should be conducted after a reactor trip or unplanned shutdown to evaluate the causes of the trip and the corrective measures implemented.

A formal communication system should exist for the transmission of orders and for the transfer of information related to the reliable and safe operation of the plant. Oral communication should be clear, concise and understandable.

More information is in IAEA (2008a) and IAEA (2011a).

23.5.5 Work authorizations

Work conducted at the plant should be planned, analysed and executed in a manner that is consistent with the requirements of plant operations both during power operation and during shutdown. A comprehensive work planning and control system shall be implemented to ensure that maintenance, testing, surveillance and inspection work is properly authorized and is carried out in accordance with established procedures. A work control process should be integrated into all work groups. By supporting this process operations will be able to better analyse risk when equipment is inoperable and decrease the time during which important equipment is not available due to inappropriate scheduling of maintenance.

The operations group has the responsibility to assist maintenance in the planning and execution of work on plant components and systems to ensure that equipment reliability and availability are maximized.

Emergent work should go through the same safety review process to evaluate risk as work in a planned schedule.

Planning of work, outages, modifications and tests should be well coordinated to ensure that the plant remains in a safe condition at all times and in accordance with the OLCs. Better planning and work control also mean that control room operations staff, maintenance technicians, system engineers, radiation protection personnel and planners are able better to coordinate their activities. The work management system should ensure that operational tasks are identified, prioritized and correctly executed. Suitable and sufficient assessments of the risks to health and safety arising from particular activities need to be carried out. The results of risk assessment need to be incorporated into the documentation for the permit to work system.

More information is in IAEA (2000b), IAEA (2008a) and IAEA (2011a).

23.5.6 Fire prevention and protection programme

The operating organization should establish and implement a comprehensive programme for fire prevention and protection to ensure that measures for all aspects of fire safety are identified, implemented, surveyed and documented throughout the entire lifetime of the plant. It is expected that the programme includes at least the following:

- Control procedures for combustible materials and ignition sources
- Inspection, maintenance, surveillance and testing of fire protection measures
- Manual fire-fighting capability
- Emergency plans, including liaison with any off-site organizations that have responsibilities in relation to fire fighting
- Integration of plant fire safety arrangements and liaison between parties involved
- Review of plant modifications to evaluate effects on fire safety
- Training in fire safety and emergency drills
- Impact of plant modifications on fire safety
- Periodic updating of the fire hazard analysis.

Responsibilities of site staff involved in the establishment, implementation and management of the programme for fire prevention and protection, including arrangements for any delegation of responsibilities, should be identified and documented. The documentation should identify the posts, specific responsibilities, authorities and chain of command for personnel involved in fire safety activities, including their relation with the plant organization. The plant management should establish an on-site group with the specific responsibility for ensuring the continued effectiveness of the fire safety arrangements.

Plant personnel engaging in activities relating to fire safety should be appropriately qualified and trained so as to have a clear understanding of their specific areas of responsibility and how these may interface with the responsibilities of other individuals, and an appreciation of the potential consequences of errors. General training relative to fire hazards, flooding, secondary effects of fires and fire zone protection should be provided to station personnel.

Periodically, drills and exercises should be conducted to confirm the fire prevention and protection programme's implementation and effectiveness. Records should be maintained of all exercises and drills and of the lessons to be learned from them. Full consultation and liaison should be maintained with any off-site organizations that have responsibilities in relation to fire fighting.

More information is in IAEA (2000b) and IAEA (2011a).

23.5.7 Management of accident conditions

Arrangements and procedures should be in place which address the actions necessary following accident conditions at a plant. The organization and administration of the direct operating group should ensure that the nuclear power plant can be controlled under accident conditions. The shift supervisor should have prompt support from the technical staff while managing accident conditions, including beyond-design-basis accident and severe accident conditions. When the conditions exceed specific limits as per the station emergency plan, an additional organization structure should be established to take over the responsibility for long-term actions to mitigate effects on the environment.

Under extreme situations an operator may be required to deviate from OLCs. The plant should have clear written directions addressing under what circumstances the OLCs may be intentionally deviated from, what permission is necessary prior to the action, and any notifications to plant staff or regulators that are required before or after the deviation occurs.

Adequate training and frequent drills using the emergency operating procedures (symptom or event oriented) and emergency plan procedures should be carried out. The members of the operating staff should receive instruction in analysis of accidents beyond the design basis and severe accidents as part of their training programme. The training of plant operators should ensure their familiarity with accidents beyond the design basis and the guidance for severe accident management.

The emergency staff and the supporting groups should be trained in performing appropriate, pre-planned actions. All the training should be repeated at sufficient intervals and reinforced through drills involving the full exercise of all emergency team members under conditions that are as realistic as possible.

More information is in IAEA (2009c) and IAEA (2011a).

23.6 Maintenance

The nuclear installations must be regularly inspected, tested and maintained in accordance with approved procedures to ensure that components, structures and systems continue to be available and to operate as intended, and that they retain their capability to meet the design objectives and the requirements of the safety analysis. The operating organization shall prepare and implement a programme of maintenance, testing, surveillance and inspection of those structures, systems and components which are important to safety. Maintenance covers in-service inspection, spare parts, materials and outage management.

More information is in IAEA (2002b) and IAEA (2011a).

23.6.1 Organization and functions

Goals, objectives and priorities of the maintenance department should be defined to be consistent with the plant policies and objectives. Maintenance strategies should be developed to address short- and long-term issues. Performance indicators should be established and used to improve performance. Effective and high-quality maintenance programmes should be encouraged by senior management. Feedback from performance results should be used in accountability reviews and in establishing goals and objectives for subsequent planning periods.

The organization and administration of the maintenance department should ensure the efficient and effective implementation and control of maintenance activities. The organization and staffing of the maintenance department, as well as the responsibilities of the different units and staff in maintenance, should be defined and communicated such that all affected personnel understand them. Succession planning should be an established practice in the maintenance department. Good coordination among different maintenance groups (mechanical, electrical, instrumentation and control, and civil), and with operations and supporting groups, should be established.

Management should demonstrate by example a continuous commitment to safety culture. They should promote safety culture and high performance standards. Their frequent presence in the field should contribute to improved job performance by the use of leadership and coaching techniques.

The organization, qualifications and number of maintenance personnel should be sufficient for the maintenance performed during the operation of the plant, the outage work to be performed by the plant's staff and the supervision of contractor's work. Contractor personnel should be subject to the same criteria as plant personnel. Good initial and continuing training should be implemented.

An emerging trend in plant maintenance and support is the increased employment of contractors to replace traditionally plant-based personnel. While this policy has financial benefits for the utility, it often comes at the expense of safety as a result of lower standards followed by contractors. The policy of relationships with contractors falls within the scope of safety culture development to ensure that the primary responsibility of the utility or plant regarding safety and monitoring is not diluted and to foster the quality factor in the contractors' activities. Emphasis must be placed on the quality and safety of work done by the contractor, who must be aware of the standards required. Contractors should receive the same attention and training in safety culture as utility staff.

More information is in IAEA (2001b), IAEA (2002b) and IAEA (2011a).

23.6.2 Maintenance facilities and equipment

Working facilities should provide sufficient space and equipment to perform maintenance activities safely and efficiently. Maintenance facilities should be clean and orderly, and maintenance tools and equipment should be maintained in good repair. Lifting, loading and transport equipment should be available and there should be provisions for auditing this type of equipment. Consideration should be given to the use of mobile lifting and transport facilities as a possible means of substantially reducing occupational exposure (for example, filter removing equipment).

Contaminated tools and equipment should be used and stored in a manner which prevents the spread of contamination. Work on contaminated equipment should be controlled in order to minimize radiation dose. Remote-controlled equipment should be available for work in high radiation areas where it has the potential to decrease radiation dose at reasonable cost.

In addition to the special equipment essential to maintenance, the plant management should provide special equipment where this could significantly reduce exposure or enhance safety and should provide adequate training in its use.

Measurement and test equipment should be controlled to assure accuracy and traceability. Chemicals and flammable material should be stored appropriately.

More information is in IAEA (2002b) and IAEA (2011a).

23.6.3 Maintenance programmes

Comprehensive programmes should optimize safe and reliable performance of plant systems and components over the lifetime of the plant. They should be established for in-service inspection, plant ageing and predictive, preventive and corrective maintenance.

These programmes should be fully integrated with plant operation and modification activities. They should be routinely reviewed and updated, as required, to take into account on-site and off-site operating experience and modifications to the plant or its operating regime. Methodologies such as probabilistic safety analysis and reliability-centred maintenance techniques should also be reviewed and updated. Risk assessment techniques can also contribute to determining maintenance and inspection requirements.

The power plant should establish a programme that takes into account the plant equipment ageing process through the various activities of operation, surveillance and maintenance.

Preventive maintenance (PM) should minimize the potential for breakdown (corrective maintenance) of important equipment by the early detection and correction of equipment degradation. PM activities should be scheduled and carried out according to a defined programme.

Predictive maintenance activities should be used to monitor the condition of installed equipment and systems where appropriate. The results of predictive maintenance activities and surveillance tests should be properly trended to permit full effectiveness of the preventive maintenance and lifetime management programmes.

The corrective maintenance programme should provide for effective reporting and timely correction of equipment degradation.

The in-service inspection programme should be established to examine systems and components of the plant for possible deterioration so as to judge whether they are acceptable for continued safe operation of the plant or whether remedial measures should be taken. The in-service inspection programme should be implemented in accordance with plant policy, regulating requirements and OLCs.

Recently in the nuclear industry, as a response to economic pressures, there are initiatives to improve efficiency and reduce costs. In the maintenance area this may lead to increases in the time periods between maintenance or inspection outages to improve capacity factors, and shortening maintenance and refuelling outage times to improve capacity factors. These initiatives should be managed in such a way that possible detrimental effects on the quality and effectiveness of the maintenance programmes are avoided.

More information is in IAEA (2002b) and IAEA (2011a).

23.6.4 Procedures, records and histories

A policy governing the use of procedures and the handling of deviations from the procedures should be implemented and communicated to staff.

Maintenance procedures and other work-related documents should identify preconditions and precautions, provide clear instructions for work to be done, and be used to ensure that maintenance is performed in accordance with the maintenance strategy, policies and programmes. The procedures should normally be prepared in cooperation with the designers, the suppliers of plant and equipment, and the personnel conducting activities for quality assurance, radiation protection and technical support. They should be technically accurate, properly verified, validated, authorized and periodically reviewed.

Priority should be given to amending and updating procedures in a timely manner. A mechanism should be implemented which enables users to feed back suggestions for the improvement of procedures.

Maintenance instructions issued to craftsmen should be compiled in accordance with quality assurance requirements and should point out the risk impact of the work on nuclear and personnel safety and identify the countermeasures to be taken and specify the post-maintenance/modification testing required. The required level of skill and methods of procedure use should be stated. Routine activities involving skills that qualified personnel usually possess may not require detailed step-by-step instructions; they should nevertheless be subject to control by means of general administrative procedures.

Human factors and 'As Low As Reasonably Achievable' (ALARA) principles should be considered in the preparation of maintenance instructions.

Maintenance history should be used to support maintenance activities, upgrade maintenance programmes, optimize equipment performance and improve equipment reliability. Appropriate arrangements should be made for orderly collection and analysis of records and production of reports on maintenance activities. Maintenance history records should be easily retrievable for reference or analysis. The use of computerized maintenance history handling would facilitate this process.

More information is in IAEA (2002b) and IAEA (2011a).

23.6.5 Conduct of maintenance work

Maintenance should be conducted in a safe and efficient manner to support plant operation. Personnel should exhibit competence and professionalism, which result in quality workmanship when performing assigned tasks. Personnel should also demonstrate a questioning attitude before, during and after the work is completed. Programmes and documentation should support this attitude.

Work should be performed in accordance with policies and procedures and be consistent with ALARA and waste minimization principles.

Maintenance personnel should be attentive to identifying plant deficiencies and responsive to correcting them with the goal of maintaining reliability and availability of equipment and systems and keeping them in optimum material condition, consistent with the design requirements.

Managers and supervisors should routinely observe maintenance activities to ensure adherence to station policies and procedures. Post-maintenance and modification testing should be systematically and thoroughly conducted.

More information is in IAEA (2002b) and IAEA (2011a).

23.6.6 Material conditions

The material condition of the plant should be maintained in such a way that its safe, reliable and efficient operation can be ensured. Plant managers and supervisors should define the required standard and conduct frequent tours of plant areas in order to confirm that high standards are maintained. Deficiencies should be identified, controlled and eliminated.

More information is in IAEA (2002b) and IAEA (2011a).

23.6.7 Work control

A comprehensive work planning and control system that considers defence in depth should be used to ensure that work activities are properly identified, prioritized, authorized, scheduled and carried out in accordance with appropriate procedures and completed in a timely manner. The work planning system should maintain high availability and reliability of important plant systems. Outage planning should be integrated into the work control process.

The effectiveness of the work control process should be monitored via appropriate indicators and corrective action taken when required. Plant defects should be tracked to completion and records kept of work performed. These records should be accessible for review when necessary. The work control process should contain an effective operational feedback system and a systematic analysis of root causes of rework or repetitive failures. Work scheduling should allocate parts, materials, resources and expertise at the appropriate time for completion of the preventive and corrective programmes and make provisions for adequate post-maintenance testing.

Improved planning and work control can increase the productivity of plant maintenance, which, in turn, can lead to a reduced maintenance backlog. This is likely to decrease the number of equipment problems with a beneficial effect in reducing the number of plant events and challenges to safety systems. Good coordination should be established among maintenance work groups, operations, other support groups and external agencies where appropriate.

More information is in IAEA (2002b) and IAEA (2011a).

23.6.8 Spare parts and materials

Materials management should ensure that necessary parts and materials, meeting established quality or design requirements, are made available and are suitable for use when needed throughout the lifetime of the plant. Regular QA audits should be conducted.

Spare parts and materials important to safety should be accompanied by documentation indicating that all requirements specified in the purchase order have been met.

Adequate storage facilities, equipment and administration should ensure correct management of materials. Suitable environmental conditions should exist and fire protection means should be provided.

More information is in IAEA (2002b) and IAEA (2011a).

23.6.9 Outage management

Outage management organization and administration should ensure the safe and effective implementation and control of maintenance activities during planned and forced outages. Outage planning and performance should take into consideration safety, quality and schedule in this order. Programmes and plans should reflect this.

Outage planning should be a continuing process involving past, next scheduled and future outages. Milestones should be determined and used to track pre-outage work. Planning should be completed as far in advance as possible as circumstances may cause the outage to begin earlier than intended.

The tasks, authorities and responsibilities of different organizational units and persons should be clearly understood. This is especially important during outage periods, when the organization may be temporarily modified. Nuclear safety during shutdown must be given careful consideration.

ALARA principles (see Section 23.6.4) and waste reduction should be embedded in programmes and planning.

More information is in IAEA (2002b) and IAEA (2011a).

23.7 Technical support

Technical support (TS) covers all on-site activities of the technical and engineering groups involved in surveillance testing, plant performance monitoring, plant modifications, reactor engineering, fuel handling, and application of plant process computers. The integration of technical support with its specialist functions into the plant organization is important in order to support and ensure the safe operation of the nuclear power plant.

More information is in IAEA (2001a), IAEA (2002a), IAEA (2002b), IAEA (2003b) and IAEA (2011a).

23.7.1 Organization and functions

The goals and objectives of TS should be written and defined within the framework of plant policies and goals and be well understood by all personnel. In those it should be clear that nuclear safety has an overriding priority. Performance indicators should be established that encourage these expectations and standards and are reported in periodic assessments.

The organization and administration of the technical support should ensure effective implementation and control of technical support activities. Effective implementation of the various technical support functions can be accomplished by having a separate section that is responsible for all such activities or by having various in-plant and off-site sections providing different support. Either method should be implemented with a well-defined organization and written assignment of responsibilities, but it should be clear that overall responsibility for safety remains with the owner of the plant. The interface between TS and other plant on-site and off-site groups should be clearly specified. Good coordination between the TS, operations and maintenance groups is of utmost importance.

The responsibilities and authorities of the technical support personnel should be clearly defined and understood by all affected personnel. The organization, qualifications and number of technical support personnel should be sufficient to accomplish assigned tasks contained in the technical support area. A system should be implemented to ensure that any person carrying out safety-related work should be suitably experienced and qualified for that function whether they are plant based or from another organization.

Design changes should be made with a full understanding of all the design information for the plant and the specifications for each system and component. Both deterministic and probabilistic assessment approaches should be used to justify and evaluate the impact of the major plant design and/or operational practices changes. The assessment process should be sound and based on safety analyses of high quality and adequate scope. Periodic safety reviews should be performed on a regular basis. Safety reviews shall address in an appropriate manner the consequences of the cumulative effects of plant ageing and plant modifications, equipment requalification, operating experience, current standards, technical developments, and organizational and management issues as well as siting aspects. The scope of the safety review shall include all safety-related aspects of an operating plant. To complement deterministic safety assessment, probabilistic safety assessment (PSA) can be used for input to the safety review to provide insights into the contributions to safety of different safety-related aspects of the plant.

The necessary knowledge of the overall plant design should be retained in a form that is practically and easily available to the operating organization over the full operating lifetime of the plant. This may be achieved by setting up a 'design authority', i.e. a design capability within the operating organization, or by having a formal external relationship with the original design organizations or their successors.

Plant management should clearly be committed to nuclear safety while providing technical support services. The integration of knowledge of the human factors into the routine day-to-day safety work, for example in the planning and implementation of a major plant modification or in the investigation of an incident, may provide a fruitful means of improving safety performance. Leadership and coaching should contribute to the improvement of safety performance. Line management should be accountable for the training and qualification of their personnel.

More information is in IAEA (2001a), IAEA (2001b), IAEA (2002a), IAEA (2002b), IAEA (2010c), IAEA (2010d), IAEA (2010e) and IAEA (2011a).

23.7.2 Surveillance programme

A comprehensive and adequately documented surveillance programme should be established and implemented to confirm that provisions for safe operation that were made in the design and checked during construction and commissioning continue to exist during the life of the plant. At the same time, the programme should confirm that safety margins are adequate and provide a high tolerance for anticipated operational occurrences, errors and malfunctions.

A surveillance test programme should verify that the plant systems and components relevant to safety are continuously ready to operate and are able to perform their safety functions as designed. Such a surveillance test programme should also detect ageing trends to prevent potential long-term degradation.

In addition a surveillance programme should detect and correct any anomalous condition before it significantly affects safety. The anomalous conditions which are of concern to the surveillance programme should include not only failures or deficiencies but also trends, analysis of which may indicate that the plant is deviating from the design intent.

The surveillance programme should be clearly documented and cross-referenced to the operating limits and conditions and safety analyses. The surveillance procedures should specify surveillance requirements and identify acceptance criteria, persons responsible for performance of surveillance activities and periodicity of each surveillance activity.

The surveillance programme should be modified if necessary in accordance with the evaluation of the data generated during surveillance and re-evaluation of the safety analysis report. The established frequency and extent of surveillance should be periodically re-evaluated to establish that they are effective in maintaining the systems, structures and components in an operational state.

More information is in IAEA (2002b) and IAEA (2011a).

23.7.3 Plant modification system

An overall plant modification programme should encompass all intended changes of structures, systems, components and process software of power plant, operational limits and conditions, instructions and procedures.

The design authority, or a responsible designer in its assigned area, should review, verify and approve (or reject) design changes to the plant. Design changes include field changes, modifications and the acceptance of non-conforming items for repair or use without modification.

A plant modification programme for permanent and temporary modifications should be established to ensure proper design, review, control, implementation and documentation of plant design changes in a timely manner. All changes requested should be reviewed, controlled, installed, tested and documented according to plant safety rules and procedures. The plant safety level after a modification should be within the design basis for the plant.

This programme should ensure that the safety significance of a modification is adequately assessed before implementation and that its impact on reliability and design configuration is also considered.

The plant modification programme should be integrated into the overall plant configuration management system that identifies documented design requirements, ensures the design is properly implemented, and controls plant changes throughout the life of the plant.

More information is in IAEA (2001a) and IAEA (2011a).

23.7.4 Reactor core management (reactor engineering)

Reactor core management should ensure the safe and optimum operation of the reactor core without compromising any OLCs based on design, safety or nuclear fuel limits. Maximum effort and priority should be assigned to maintaining fuel integrity. The core management programme should also provide tools to control core management and ensure that only approved fuel is loaded into the core.

The core management programme should include appropriate numerical methods and techniques to predict reactor behaviour during operation so as to ensure that the reactor will be operated within OLCs. The core parameters should be monitored, trended and evaluated in order to detect abnormal behaviour and ensure that actual core performance is consistent with core design requirements. To ensure that fuel cladding integrity is maintained under all core operating conditions, radiochemistry data that are indicative of fuel cladding integrity should be systematically monitored and analysed for trends. An adequate fuel failure contingency plan or policy should be established and implemented to ensure that corrective actions for failed fuel are taken.

A core management programme should also include the surveillance activities for the early detection of any deterioration that could result in an unsafe condition in the reactor core. The personnel involved in the core management should be well qualified, have clear responsibilities and authorities and be readily available to support plant operations during all modes of operation.

More information is in IAEA (2002a) and IAEA (2011a).

23.7.5 Handling of fuel and core components

The handling programme for fuel and other core components should provide measures to prevent damage to the nuclear fuel and to prevent inadvertent criticality and loss of appropriate cooling when fuel assemblies are being transported, stored or manipulated. For purposes of radiological protection, precautions to be taken in handling unloaded fuel, core components and materials and any disassembly operations should be specified in the procedures. The handling programme should also ensure that all procedures and controls adequately reflect radiation protection requirements and plant policies for ALARA considerations (see Section 23.6.4).

The comprehensive fuel handling programme should include receipt, transfer, inspection and storage of nuclear fuel. Fuel handling planning should accomplish fuel loading and unloading safely in accordance with a core management programme as well as safe storage, handling and preparation for dispatch of the irradiated fuel. Fuel elements should be traced by means of an appropriate system to maintain a thorough fuel inventory and history. Each core component should be adequately identified and a record should be kept of its core location, orientation within the core, out-of-core storage position and other pertinent information so that an irradiation history of the component is available.

More information is in IAEA (2002a) and IAEA (2011a).

23.7.6 Computer-based systems important to safety

A programme for utilization of computer-based systems should be established and implemented to support and verify the safe operation of the plant. Utilization of computer-based systems may vary greatly between different plants. The programme for utilization should therefore clearly define the categorization of the applications in terms of their safety significance. This section of the guidelines refers (if not stated specifically) to both safety systems and safety-related systems.

Organizational responsibilities for computer-based applications should be well defined and meet the needs for ensuring safe plant operation. This includes well-organized documentation and provisions for emergency recovery of failed software applications.

To ensure the appropriate operation of different computer-based systems according to their design functions, a relevant section should be established in the quality assurance programme.

More information is in IAEA (2004) and IAEA (2011a).

23.8 Operational experience feedback (OEF)

Operating experience feedback (OEF) is a key element in maintaining and improving the safety of nuclear installations operations. The International Reporting System ((IRS) as a worldwide system is designed to complement national schemes in this regard. Information reported is assessed, analysed and fed back to operators for their use in preventing similar occurrences. The ultimate objective is to enhance the safety of nuclear facilities by reducing the frequency and severity of safety-significant events at nuclear facilities worldwide. Nevertheless international OEF is an area where the nuclear industry should have more efficient programmes. Safety-related events continue to happen and subsequent analysis reveals that there had been similar events in the past at other facilities, and/or that precursors or latent root causes of the event at the given facility had been in existence but had not been dealt with efficiently.

A well-implemented operational experience (OE) programme is characterized by the following features:

- Management aligns the organization to effectively implement the OE programme in order that plant safety and reliability are improved.
- OE is reported in a timely manner to reduce the potential for recurring events in-house and in the industry.
- Sources of OE are considered in the OE programme to improve plant safety and reliability from the lessons learned.

- OE information is appropriately screened to select and prioritize those items requiring further investigation.
- Analysis is performed on appropriate events, depending on their severity or frequency, to ensure root causes and corrective actions are identified.
- Corrective actions are defined, prioritized, scheduled and followed up to ensure effective implementation and effective improvement of plant safety and reliability.
- OE information is used throughout the plant to effectively improve plant safety and reliability.
- OE information is analysed and trended, and the results are used to improve plant safety and reliability.
- Assessments and indicators are effectively used to review and monitor the plant performance and the effectiveness of the OE programme.

More information is in IAEA (2006b), IAEA (2008b) and IAEA (2011a).

23.8.1 Management, organization and functions of the OE programme

A programme of OE should be in place, covering all areas of the OE feedback process. Effective use of OE is part of the safety culture. Management is committed and involved in promoting and reinforcing the use of OE to improve plant safety and reliability. Policy, goals, objectives and management expectations are clearly defined and communicated. The programme is developed in procedures for the management of the internal OE, including low-level events and near misses, external OE, periodic assessment of OE activities and programme review.

Duties, responsibilities, authorities and lines of communication within the plant organization are clearly defined and understood. Duties, responsibilities, authorities, lines of communication and interfaces of corporate organizations as well as other external organizations in the OE process are clearly defined and understood. Tools such as methods, criteria and appropriate training are provided to perform the tasks of the OE feedback process. Adequate resources are allocated for the OE programme including coordination. A group is identified to manage the process.

Active participation in OE activities is implemented throughout the plant in a blame-free atmosphere. Supervisors and managers actively reinforce effective use of OE information by personnel.

Personnel are held accountable for effective analysis and timely implementation of lessons learned from OE information. Comprehensive monitoring of the tasks carried out in the OE process is performed for compliance with the targets defined.

The effectiveness of the OE process is monitored regularly. A clear feedback process exists in which the results of the monitoring are transmitted to the responsible groups affected by the results.

More information is in IAEA (2004) and IAEA (2011a).

23.8.2 Reporting of operating experience

OE is identified and reported in a timely manner according to well-established criteria and procedures. Problem identification and reporting is strongly encouraged and reinforced at all levels in the organization.

Significant events, minor events, low-level events, near misses and potential problems are identified and reported, including equipment failures, human performance problems, procedure deficiencies and documentation inconsistencies.

Dissemination of OE to plant personnel and dissemination of significant experience to other nuclear power plants are promptly performed.

More information is in IAEA (2004) and IAEA (2011a).

23.8.3 Sources of operating experience

Sources of industry operating information are identified, and access to these sources is formally established and systematically screened. These sources include organizations (IAEA, NEA, WANO, INPO, national Regulatory Body, owners' groups, vendors and manufacturers, engineering designer) and publications (IRS, SER, SOER; national Regulatory Body generic letters, bulletins, notices; vendors, manufacturers and engineering designer problem information; utilities and industry event reports). Sources of OE include good practices as a source of improvement.

The International Reporting System (IRS) for operating experience is an international system jointly operated by IAEA and NEA, through which 31 participating countries exchange experience to improve the safety of nuclear power plants by submitting event reports on unusual events considered important for safety. More than 3500 reports are in the IRS. In addition the periodic reports are published every three years. These reports highlight important lessons learned based on a review of the approximately 200 event reports received from the participating countries over a period of three years (see IAEA/NEA, 2010b).

Sources of in-house OE are identified, and information from and access to these sources are formally established and systematically screened. These sources include areas such as significant events, low-level events and near misses, quality reports, reports and data from operation activities, maintenance testing and in-service inspection, surveillance reports, results from plant-specific safety assessments, training feedback, no-blame reporting programme, and performance indicators.

More information is in IAEA (2004) and IAEA (2011a).

23.8.4 Screening of operating experience information

OE information is appropriately screened, to select and prioritize the information for further investigation. Screening criteria for in-house and industry OE are clearly established and the criteria for the subsequent level of investigation and distribution are defined.

The screening is performed in a systematic and timely manner. The sources for screening and their corresponding frequency of screening are defined. Screening is performed by individuals with a broad knowledge of plant operations or by a multidisciplinary group.

More information is in IAEA (2004) and IAEA (2011a).

23.8.5 Analysis

Analysis is performed on the selected events in accordance with their level of safety significance, severity and frequency to ensure that root causes and corrective actions are identified. Criteria for performing a full root-cause analysis, a simplified analysis, and a trending analysis are clearly defined in the OE programme, and procedures are developed.

For significant in-house events, including scrams, plant transients and important human performance and equipment problems, a rigorous investigation with full root-cause analysis is performed, including causal factors, generic implications, and discrepancies between expected and actual plant responses and/or personnel actions.

For low-level events and near misses, minor events, no-consequence events or any other likely useful error information and potential problems, the level of analysis required is clearly defined such that generic implications, precursors of declining performance and root causes of adverse trends can be identified. Determination of corrective actions allows the correction of latent weaknesses and the prevention of recurrence.

Personnel who have appropriate knowledge, experience and skills perform investigations and analysis. Event participants are involved in developing and implementing corrective actions, as necessary.

Investigation of events is initiated promptly to preserve information and physical evidence and to interview participants while the events are fresh in their memories. Investigations are carried out in a timely manner.

Investigation and analysis take account of previous similar events and precursors from both internal and external sources. Investigations and analysis are subject to objective review to ensure that root causes have been identified, which are then addressed by effective corrective actions.

More information is in IAEA (2004) and IAEA (2011a).

23.8.6 Corrective actions

The results of OE reviews and analysis are used to identify corrective actions. Corrective actions address fundamental causes of problems, rather than the symptoms, to avoid recurrence of events.

Corrective actions are prioritized, scheduled for implementation, and effectively implemented. Dates for actions are commensurate with the importance of the item, station priorities, and the consideration of preventing the recurrence. Operating shift crews are promptly briefed on events and compensatory measures are taken to prevent recurrence.

Corrective actions are tracked for completion to verify their implementation. The status and effectiveness of corrective actions are periodically reviewed. Management receives feedback on the review results.

More information is in IAEA (2004) and IAEA (2011a).

23.8.7 Use of operating experience (OE)

OE information is used throughout the station. Personnel are aware of management expectations to use OE information. OE information is easily accessible to station personnel. Personnel are aware and knowledgeable on how to access it.

Use of OE in personnel work activities (i.e. pre-job briefings and pre-evolution briefings, work planning, shift briefings, etc.) is carried out to remind the personnel involved of lessons learned and precautions from OE, to enhance the personnel alertness and to reduce risks.

OE information is used in training. It is compiled in training modules for operators' simulator training and in training of plant personnel in other areas.

More information is in IAEA (2004) and IAEA (2011a).

23.8.8 Database and trending of operating experience

Databases related to events, deficiencies, anomalies and deviations are established to facilitate an integral view and analysis of OE from the point of view of organizational aspects, human factors, equipment failures, work management and maintenance deviation reports. For significant events, low-level events (minor events) and near misses (non-consequential events, potential problems), database trending system representations (trending parameters) are established to provide transparent data presentation that facilitates diagnosis of monitored performance, identification of patterns, identification of abnormal trends, identification of recurrences, quick plant management overview and action focus. Trend analysis is carried out on a regular basis and the results of analysis are reported to management. Actions are taken to correct identified adverse trends with potential for undesirable consequences.

More information is in IAEA (2004) and IAEA (2011a).

23.8.9 Assessments and indicators of operating experience

Self-assessments and independent evaluation are periodically performed to determine the effectiveness of the OE programme and the effective use of OE information. Self-assessment evaluates all steps of the OE process. Management receives feedback on the self-assessment results. The results of self-assessment are used to identify weaknesses in the OE programme and to make the necessary improvements.

Indicators are used to monitor the safety performance of the plant. The trends of indicators are evaluated during self-assessment. Examples of these indicators are recurrent unavailability of safety systems, industrial safety events, reactor scrams, volume of low-level waste, and radiation doses.

Indicators are used to track the effectiveness of the OE programme. Examples of these indicators are average time for initial screening of OE documents, number and age of reports awaiting evaluation, number and age of corrective actions awaiting implementation, recurrent events and root causes, reworks, and the ratio of events detected through surveillance and quality programmes versus operational failures or degradation in service.

Benchmarking with industry indicators is performed and results of the comparison are considered to determine opportunities for improvement.

More information is in IAEA (2004) and IAEA (2011a).

23.9 Radiation protection

The operating organization shall establish and implement a radiation protection programme. The radiation protection (RP) function in the operating organization shall have sufficient independence and resources to be able to enforce and to advise on the radiation protection programme. The radiation protection regime established and implemented by the operating organization at a nuclear power plant should ensure that in all operational states doses due to exposure to ionizing radiation in the plant or due to any planned releases of radioactive material from the plant are kept below prescribed limits and ALARA. Controls for RP during operation of the plant, including the management of radioactive effluents and waste arising in the plant, should be directed not only to protecting workers and members of the public from radiation exposure, but also to preventing or reducing potential exposures and mitigating their potential consequences.

More information is in IAEA (1996), IAEA (1999a), IAEA (1999b), IAEA (1999c) IAEA (2002c), IAEA (2006e) and IAEA (2011a).

23.9.1 Organization and functions

The RP goals and objectives should be clearly defined in the safety policies of the operating organization and communicated to the personnel and the management of the power plant. To achieve these goals and objectives, a well-structured RP programme should be established and implemented. The programme should be documented in the plant policies and procedures and shall meet the requirements of the International Basic Safety Standards for Protection against Ionizing Radiation and for the Safety of Radiation Sources (BSS). The management should ensure that the RP policies and procedures are well understood by the plant's personnel. The RP programme should be clearly oriented to the achievement of a level of performance in RP that is well above minimum regulatory requirements.

Effective implementation of the RP programme should be supported by establishing written procedures requiring high performance in RP, periodically monitoring and assessing performance, and holding personnel accountable for their performance. Performance indicators should be established that encourage the management expectations and standards and are reported in periodic assessments.

The RP function in the operating organization shall have sufficient independence and resources to enforce and give advice on RP regulations, standards and procedures and safe working practices. Sufficient staff, equipment and funding should be provided to successfully implement the RP programme. An independent radiation protection group should be established, which has the authority to enforce RP regulations, standards, procedures, safe working practices and appropriate health physics surveillance. Succession planning should be an established practice in the RP group. The RP manager at the plant should have direct access to the plant's manager on the matters relating to the radiation protection. The RP organization should be well defined and understood, including the interfaces with other plant groups.

All levels of management and workers should be committed to RP requirements and safe work practices within their level of responsibility. The RP group as well as the workers and management should be trained and qualified in RP issues to a level appropriate to their responsibilities. All personnel of the plant should be aware of radiological hazards and of necessary protective measures.

The RP programme shall provide for health surveillance of site personnel who may be occupationally exposed to radiation to ascertain their physical fitness and to give advice in cases of accidental overexposure.

The operating organization shall verify, by means of surveillance, inspections and audits, that the RP programme is being correctly implemented and that its objectives are being met, and shall undertake corrective actions if necessary. The programme shall be reviewed and updated in the light of experience.

The principal objective of incorporating QA principles into RP should be to improve safety by establishing confidence in the results of RP. Additional benefits should be the strengthening of efficiency and effectiveness by establishing a system for improving RP based on the use of relevant experience (lessons learned), the identification and prompt correction of deficiencies, and the monitoring of performance.

More information is in IAEA (1996), IAEA (1999a), IAEA (1999b), IAEA (1999c), IAEA (2002c), IAEA (2006e) and IAEA (2011a).

23.9.2 Radiation work control

Exposure to sources of external and internal radiation at nuclear power plants should be reduced to such dose levels that are as low as reasonably achievable (ALARA). This principle should apply both to individual and to collective doses. The responsibility for optimizing occupational exposure should rest both with management of different levels and with the RP group. Work in controlled areas should be authorized in accordance with appropriate procedures. Control of all entrances to and exits from radiological areas should be established and maintained. A programme for monitoring of radiological conditions should be established for designated areas.

More information is in IAEA (1996), IAEA (1999a), IAEA (1999b), IAEA (1999c), IAEA (2002c), IAEA (2006e) and IAEA (2011a).

23.9.3 Control of occupational exposure

The occupational exposure at the power plant should be so controlled that the dose limits recommended by the International Commission on Radiological Protection (ICRP) and required by the IAEA Safety Standards are not exceeded. These limits shall be transported to the national regulations. The optimization of protection and safety measures, or the application of the ALARA principle (to keep doses as low as reasonably achievable, economic and social factors being taken into account), should be carried out. In examining working procedures and activities, the reduction of doses should be given the highest priority. A hierarchy of control measures should be taken into account in optimization. Firstly, removal or reduction in intensity of the source of radiation should be considered. Only after this has been done should the use of engineering means to reduce doses be considered. The use of systems of work should then be considered and, lastly, the use of personal protective equipment.

Dose monitoring of individuals and management of dose records should comply with requirements established by the regulatory authority and should be consistent with the applicable recommendations of ICRP and IAEA. Exposures related to working in controlled areas should be individually monitored and recorded in order to ensure that the ALARA principle is met and that regulatory limits are not exceeded. In situations where significant concentrations of airborne activity are anticipated, appropriate internal dosimetry should be available, including whole-body counters. Provisions for indirect monitoring as an additional method for evaluating internal exposure should exist.

More information is in IAEA (1996), IAEA (1999a), IAEA (1999b), IAEA (1999c), IAEA (2002c), IAEA (2006e) and IAEA (2011a).

23.9.4 Radiation protection instrumentation, protective clothing and facilities

Adequate radiological instrumentation, protective clothing, facilities and equipment for both normal and emergency situations should be provided as part of the RP programme. The equipment and devices used to obtain radiological measurements and doses should be calibrated, maintained and used so that results are accurately determined. An adequate quantity of protective equipment and clothing should be available.

More information is in IAEA (1996), IAEA (2002c) and IAEA (2011a).

23.9.5 Radioactive waste management and discharges

The generation of radioactive waste should be kept to the minimum practicable in terms of both activity and volume, by appropriate operating practices. The operating organization should establish and implement a programme to safely manage radioactive waste and monitor and control discharges of radioactive effluents. The operating organization should perform a safety analysis for radioactive discharges, which demonstrates that the assessed radiological impacts and doses to the general public are kept as low as reasonably achievable. Any authorized discharge limits should be included in the OLCs. Radioactive waste and effluent releases should be documented as required and an environmental monitoring programme should be in place.

More information is in IAEA (2002c), IAEA (2005b), IAEA (2006e) and IAEA (2011a).

23.9.6 Radiation protection support during emergencies

The programme for RP support during emergencies should be comprehensive and serve the purpose of optimizing both worker exposure and the exposure of the general public to the extent consistent with emergency conditions.

Procedures and qualified personnel should be in place to provide technical and operational support during emergency interventions. Periodic training and practical exercises should be undertaken to ensure an effective response in the event of an emergency.

More information is in IAEA (2000c), IAEA (2002e) and IAEA (2011a).

23.10 Chemistry

Chemistry involves activities of chemical treatment to maintain the integrity of the barriers retaining radioactivity, including fuel cladding and primary circuit. The chemistry activities have a direct impact in limiting all kinds of corrosion processes causing either direct breach of safety barriers or weakening of them so that failure could occur during a transient.

In addition the chemical treatment includes consideration of its effects on the out-of-core radiation fields that in turn influence radiation doses to which the workers are exposed. Plant radiochemistry is included in the chemistry considerations for the purpose of this chapter.

More information is in IAEA (2011a) and IAEA (2011b).

23.10.1 Organization and functions

The operating organization should establish a chemistry policy for the nuclear power plant. The policy should state the goals and objectives of the chemistry programme and the expectations of the management concerning the implementation of this programme at the plant. Performance indicators should be established that encourage these expectations and are reported in periodic assessments.

A specific chemistry group should be established at the plant to implement the chemistry control programme. The organization of the chemistry group should contribute to safe operation, define responsibilities and establish lines of communication inside and outside the group. The position of this group in the organization should reflect its relevance. The interfaces between the chemistry group and other groups should be clearly specified, especially as regards allocation of authorities. The chemistry group should be consulted when issues affecting chemistry are being addressed. The qualifications and number of chemistry personnel should be sufficient for assigned responsibilities and to support all plant operations. Succession planning should be an established practice in the chemistry group.

The chemistry group's expectations, goals and objectives should be derived from the plant policies and objectives and defined in line with vendor recommendations and international good practice. They should be well understood by the chemistry personnel.

The monitoring of the chemistry group's performance and its programmes should include self-assessment of managerial processes and work performance.

More information is in IAEA (2011a) and IAEA (2011b).

23.10.2 Chemistry control in plant systems

The plant should have established and implemented a comprehensive chemistry control programme. This programme should be implemented by clear procedures and monitored by adequate performance indicators. The plant staff concerned should have a good understanding of the programme, procedures and indicators.

The chemical treatment should take into account plant material concept, and any change in plant material concept should be evaluated by the chemistry group.

The generation and transport of radioactive products within the primary system should be understood, controlled and minimized.

Some results of the chemistry analyses are issued through computer software. Checks should be made that this software is kept up-to-date.

Chemical treatments should be optimized with respect to environmental and radwaste aspects. There should be a written concept of such optimization along with procedures to support implementation of this concept.

More information is in IAEA (2011a) and IAEA (2011b).

23.10.3 Chemistry surveillance programme

The chemistry surveillance programme should include the monitoring, sampling and trending of chemistry and radiochemistry parameters at specified frequencies to ensure the timely detection and correction of abnormal or unacceptable trends and conditions. The chemistry surveillance programme should reflect chemistry specifications for all phases of plant operation, including shutdown periods and when systems are taken out of operation for prolonged periods.

Procedures for analysis and measurement should be available and well understood by the personnel of the chemistry group. Personnel doing the analysis should be technically qualified and their performance periodically assessed. Analysis techniques should be appropriate and safe and evaluated results should be transmitted in a timely manner to the appropriate operational personnel. The chemistry data should be constantly evaluated to identify chemistry control problems and analytical errors and to remove the deficiencies.

Checks should be made that the responsibilities for QA are defined and the QA programme is implemented and evaluated.

More information is in IAEA (2011a) and IAEA (2011b).

23.10.4 Chemistry operational history

The results of analysis and investigations must be adequately trended, evaluated and reported. Records should be available and easily retrievable. Lessons and experiences from previous events and history, including from other plants, should be considered in the plant chemistry.

More information is in IAEA (2011a) and IAEA (2011b).

23.10.5 Laboratories, equipment and instruments

The laboratories should have adequate space, supplies and equipment. The sampling systems should be reliable and safe for use, including post accident sampling systems. Necessary and adequate instruments for performing the analysis should be available and calibrated.

More information is in IAEA (2011a) and IAEA (2011b).

23.10.6 Quality control of operational chemicals and other substances

The purity and nature of chemicals and other substances that might have an impact on safety-related systems should be specified and controlled. Before being used the specified values should be verified by certification or by chemical analysis.

More information is in IAEA (2011a) and IAEA (2011b).

23.11 Emergency planning and preparedness

Emergency preparedness is the ability to take actions that will effectively mitigate the consequences of an emergency for human health and safety, quality of life, property and the environment. This section refers to emergency planning and preparedness both on site (operator responsibility) and off site (mostly local and state authorities' responsibility).

The practical goals of emergency response in a nuclear or radiological emergency are:

- To regain control of the situation
- To prevent or mitigate consequences at the site
- To prevent the occurrence of deterministic health effects in workers and the public
- To render first aid and manage the treatment of radiation injuries
- To prevent, to the extent practicable, the occurrence of stochastic health effects in the population
- To prevent, to the extent practicable, the occurrence of adverse non-radiological effects on individuals and among the population
- To protect, to the extent practicable, the environment and property
- To prepare, to the extent practicable, for the resumption of normal social and economic activity.

The goals of emergency response are most likely to be achieved by having a sound programme for emergency preparedness in place as part of the infrastructure for protection and safety. The practical goal of emergency preparedness is to ensure that arrangements are in place for a timely, managed, controlled, coordinated and effective response both on site and off site (at the local, regional, national and international levels) to an emergency.

For that purpose, an emergency preparedness programme is necessary that includes national, local and on-site response organizations. In a consolidated approach, the elements to be evaluated may be addressed by the operator, the local authorities or the national authorities, or by a combination thereof, so long as the arrangements are well coordinated. Weaknesses at one level could be compensated at another.

More information is in IAEA (2002e), IAEA (2007) and IAEA (2011a).

23.11.1 Emergency programme

Arrangements including clearly assigned authorities and responsibilities, organization, coordination, personnel, plans, procedures, facilities, equipment and training should be in place to provide reasonable assurance of an effective response in the case of any nuclear or radiological emergency at the site that meets the practical goals of emergency response.

An effective administrative framework should be available for the planning, implementation, coordination and control of emergency preparedness activities. This framework should be well documented, defining responsibilities and authorities, and should consider appropriately the requirements of the regulatory authority.

The operating organization's policy should ensure that all emergency preparedness activities at the plant are properly organized and are integrated with those of the operating organization's headquarters, the relevant emergency services and the local and national authorities, with due consideration to interface implications. Authorities and responsibilities should be well established and clear among all organizations involved.

The organization should ensure that adequate human and financial resources are allocated, that critical response functions are covered and that the state of preparedness is properly maintained, regularly tested and updated. All emergency planning and preparedness activities should be properly covered by the QA programme.

A close and cooperative relationship should be maintained between on- and off-site response organizations.

The response organizations periodically should conduct a review in order to ensure that all the events (including those of very low probability) that could necessitate an emergency response are addressed by the emergency arrangements. This includes a review and appropriate revision of the emergency arrangements before any revisions to existing operations or new operations are commenced on the site or nearby that may result in events warranting an emergency response.

More information is in IAEA (2002e), IAEA (2007) and IAEA (2011a).

23.11.2 Response functions

The emergency preparedness arrangements in place should provide for reasonable assurance that the response functions discussed in this section can be performed effectively during an emergency.

More information is in IAEA (2002e), IAEA (2007) and IAEA (2011a).

23.11.3 Emergency plans and organization

Approved emergency plans should clearly allocate responsibilities and provide a basis for development of procedures, training and other arrangements that provide for a coordinated response by the operating organization and other authorities.

The emergency plans should include arrangements for emergencies involving a combination of non-nuclear and nuclear hazards and response of conventional response organizations such as law enforcement. These plans should be reviewed regularly taking into consideration the feedback from drills and exercises and to consider any revisions to facility operations, terrorist threat situations, or activities/conditions in the area that may impact on the potential emergencies to be addressed or the response.

More information is in IAEA (2002e), IAEA (2007) and IAEA (2011a).

23.11.4 Emergency procedures

Procedures and analytical tools should be available and validated and should provide detailed guidance for the rapid and effective implementation of the response functions mentioned in Section 23.11.2. On-site procedures should be linked with the plant document and records management system.

More information is in IAEA (2002e), IAEA (2007) and IAEA (2011a).

23.11.5 Emergency response facilities

Facilities should be provided for adequate on-site and off-site emergency response with appropriate communications and equipment that can be brought into operation without delay in the event of an emergency. These should include centres from which the on-site and off-site emergency response can be managed, as well as means for assessment of the plant status and radiological conditions and for implementation of any necessary response actions or protective measures. In addition, special facilities for the protection of the personnel and the public, e.g. gathering points and medical centres, should be available.

More information is in IAEA (2002e), IAEA (2007) and IAEA (2011a).

23.11.6 Emergency equipment and resources

Adequate emergency equipment and resources, communication systems, documentation (such as procedures, checklists, telephone numbers and manuals) should be available where needed to properly initiate and support the emergency response actions described in Section 23.11.4. Necessary data transfer and communication should also be available.

Instruments, tools, equipment, documentation and communication systems to be used in emergencies should be appropriate and maintained in good operating condition, in such a manner that they are unlikely to be made unavailable by the postulated emergency and environmental conditions. Equipment, communications, vehicles, etc., should be regularly checked and tested.

More information is in IAEA (2002e), IAEA (2007) and IAEA (2011a).

23.11.7 Training, drills and exercises

A comprehensive, documented training programme should be provided for developing and maintaining the necessary knowledge, skills and physical ability required for all persons having duties under the emergency plans, to enable them to respond correctly and efficiently in the event of an emergency. A programme should also be provided for general employee training of on-site personnel. Similar training, or at least a well-structured information briefing, should be provided to plant visitors.

A programme of periodic drills and exercises should be set up to reinforce the training and assess the effectiveness of the emergency response capability. The programme should include periodic, comprehensive and integrated on-site and off-site exercises aimed at assessing the coordinated response of all emergency response organizations and should include evaluation of exercises for experience feedback.

More information is in IAEA (2002e), IAEA (2007) and IAEA (2011a).

23.11.8 Quality assurance

A quality assurance and maintenance programme should be in place that ensures a high degree of availability and reliability of all plans, procedures, supplies, equipment, communication systems and facilities necessary to perform the specified functions in an emergency.

More information is in IAEA (2002b), IAEA (2002e) and IAEA (2011a).

23.12 Operational Safety Review Team (OSART)

The Operational Safety Review Team (OSART) is one of the more prominent IAEA efforts that help countries to achieve higher levels of safety. The OSART programme is the main approach to providing for better and wider application of the safety standards. The primary function of the OSART programme is to assess the activities of and provide advice to the host plant on the basis of the IAEA's safety standards and to introduce the OSART methodology for the host plant to establish or improve its own self-assessment programme.

The OSART programme broadly covers nine operational areas: management, organization and administration; training and qualification; operations; maintenance; technical support; operating experience; radiation protection; chemistry; and emergency planning and preparedness. A recent enhancement of the OSART review is the addition of new optional modules concerning the transition from operation to decommissioning, accident management, long-term operation and application of probabilistic safety assessment.

In OSART missions, the IAEA coordinates internationally based teams of experts who conduct reviews of operational safety performance at nuclear power plants. The reach of OSART is expansive: to date, OSARTs have visited every major type of nuclear reactor, and over 160 reviews have been conducted since the programme's inception in 1982. NPPs are in operation in 29 countries. The share of electricity production is shown in

Fig. 23.4. With the accumulation of its results, OSART has also been greatly appreciated for providing the opportunity for mutual learning and sharing of knowledge and experience, such as good practices and lessons learned, among team members who are drawn from different Member States and host plant personnel.

The OSART review is a process that begins with a request from a country for a safety review, and can occur in three stages. A preparatory visit is conducted about 12 months before the safety review mission. The main goal is to make all necessary arrangements and help the plant for mission preparation.

Safety review missions consist of a regular OSART mission, offering an in-depth assessment of operational safety. Human performance issues and recognized operational issues are assessed in an integrated way. These regular OSART missions are concluded by follow-up visits, which take place approximately 12–18 months after an OSART mission. The follow-up provides an independent assessment of progress in the resolution of issues identified in the OSART mission.

In addition, pre-OSART missions are conducted during the construction and commissioning phase of a plant's life. These missions help ensure effective preparations for commissioning and operations.

Once the IAEA receives the review request, it begins to assemble a team of 10–12 experts to undertake the mission. The team is comprised of specialists from around the globe who have senior-level nuclear operator experience, and each team member is assigned an area of focus during the mission.

The bulk of the work for a regular OSART is carried out during an intense three weeks of review at the plant, whereby OSART mission staff conduct interviews with plant staff, observe plant workers, and analyse documents related to plant operation.

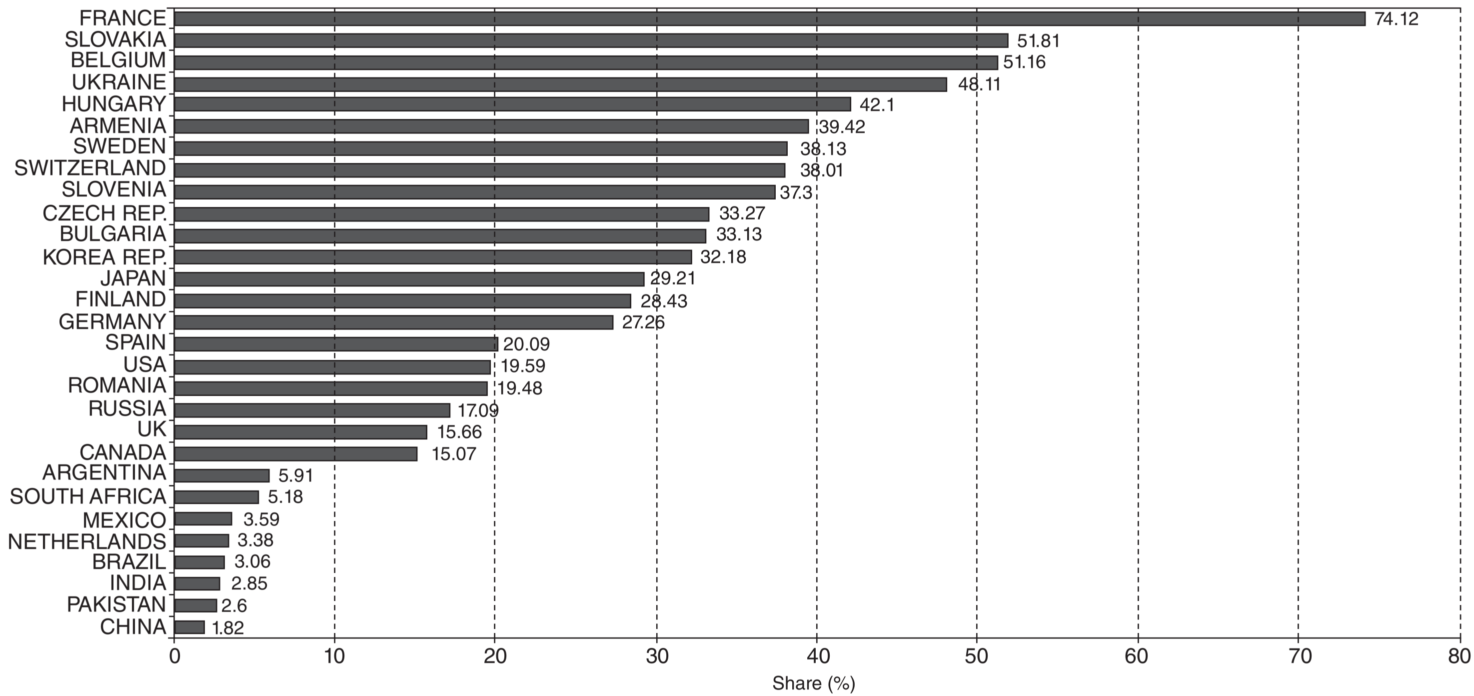
Rather than examining the plant's physical design, OSART team members are tasked with studying the operation of the plant and the performance of the plant's management and staff. OSART focuses more on the human aspect of a nuclear plant rather than the technology behind its operation.

OSART reviews are based wholly upon IAEA Safety Standards, which are established to give guidance to Member States on the many aspects of the safety of nuclear installations.

All necessary details are described in OSART Guidelines (IAEA, 2005a).

23.13 Sources of further information and advice

The International Atomic Energy Agency, in addition to its Safety Standards, produces many different types of publications relevant to operational safety. They are available on the IAEA website at http://www.iaea.org/publications/.



Note: The nuclear share in Taiwan, China, was 19.3%.

23.4 Nuclear share in electricity generation in 2010.

The Nuclear Energy Agency (NEA) is a specialized agency within the Organization for Economic Co-operation and Development (OECD), www.oecd-nea.org. The mission of the NEA is to assist its Member Countries in maintaining and further developing, through international cooperation, the scientific, technological and legal bases required for the safe, environmentally friendly and economical use of nuclear energy for peaceful purposes. To achieve this, the NEA works as a forum for sharing information and experience and promoting international cooperation, a centre of excellence which helps Member Countries to pool and maintain their technical expertise, and a vehicle for facilitating policy analyses and developing consensus based on its technical work. In the area of nuclear safety the main goal is to assist Member Countries in ensuring high standards of safety in the use of nuclear energy, by supporting the development of effective and efficient regulation and oversight of nuclear installations, and by helping to maintain and advance the scientific and technological knowledge base. Reports produced by the Committee on Nuclear Regulatory Activities (CNRA) and the Committee on the Safety of Nuclear Installations (CSNI) are of special relevance to operational safety. The Radioactive Waste Management Committee (RWMC), the Committee on Radiological Protection and Public Health (CRPPH) and the Nuclear Law Committee (NLC) also produce valuable reports, some of them in cooperation with the IAEA. Some of the documents are available on the website; others are only available to Member Countries.

The mission of the World Association of Nuclear Operators (WANO), www.wano.info, is to maximize the safety and reliability of NPPs worldwide by working together through mutual support, exchange of information, and emulation of best practices. Some publications are on INPO's website; most of them are, however, available only for members.

The mission of the Institute of Nuclear Power Operations (INPO), www.inpo.info, is to promote the highest levels of safety and reliability – to promote excellence – in the operation of commercial nuclear power plants.

There are other international organizations such as the International Labour Organization (ILO), which provides guidance on workers' health and safety, the International Commission on Radiological Protection (ICRP), which publishes recommendations on dose limits, and the World Health Organization (WHO), among others, which could be included as providers of information for the protection of workers in the operating plant.

The reader can find much additional information on the websites of nuclear vendors, nuclear utilities and regulatory organizations in different countries.

23.14 References and further reading

IAEA (1991), *Safety Culture*, INSAG-4, IAEA, Vienna.

IAEA (1996), *International Basic Safety Standards for Protection against Ionizing Radiation and for the Safety of Radiation Sources*, Safety Series no. 115, IAEA, Vienna.

IAEA (1999a), *Occupational Radiation Protection*, Safety Guide, Safety Standards Series, RS-G-1.1, IAEA, Vienna.

IAEA (1999b), *Assessment of Occupational Exposure Due to Intakes of Radionuclides*, Safety Guide, Safety Standards Series, RS-G-1.2, IAEA, Vienna.

IAEA (1999c), *Assessment of Occupational Exposure Due to External Sources of Radiation*, Safety Guide, Safety Standards Series, RS-G-1.3, IAEA, Vienna.

IAEA (2000a), *Operational Safety Performance Indicators for Nuclear Power Plants*, TECDOC 1141, IAEA, Vienna.

IAEA (2000b), *Fire Safety in the Operation of Nuclear Power Plants*, Safety Guide, Safety Standards Series, NS-G-2.1, IAEA, Vienna.

IAEA (2000c), *Operational Limits and Conditions and Operating Procedures for Nuclear Power Plants*, Safety Guide, Safety Standards Series, NS-G-2.2, IAEA, Vienna.

IAEA (2001a), *Modifications to Nuclear Power Plants*, Safety Guide, Safety Standards Series, NS-G-2.3, IAEA, Vienna.

IAEA (2001b), *The Operating Organization for Nuclear Power Plants*, Safety Guide, Safety Standards Series, NS-G-2.4, IAEA, Vienna.

IAEA (2002a), *Core Management and Fuel Handling for Nuclear Power Plants*, Safety Guide, Safety Standards Series, NS-G-2.5, IAEA, Vienna.

IAEA (2002b), *Maintenance, Surveillance and In-service Inspection in Nuclear Power Plants*, Safety Guide, Safety Standards Series, NS-G-2.6, IAEA, Vienna.

IAEA (2002c), *Radiation Protection and Radioactive Waste Management in the Operation of Nuclear Power Plants*, Safety Guide, Safety Standards Series, NS-G-2.7, IAEA, Vienna.

IAEA (2002d), *Recruitment, Qualification and Training of Personnel for Nuclear Power Plants*, Safety Guide, Safety Standards Series, NS-G-2.8, IAEA, Vienna.

IAEA (2002e), *Preparedness and Response for a Nuclear or Radiological Emergency*, Safety Requirements, Safety Standards Series, GS-R-2, IAEA, Vienna.

IAEA (2002f), *Key Practical Issues in Strengthening Safety Culture*, INSAG-15, IAEA, Vienna.

IAEA (2003a), *Periodic Safety Review of Nuclear Power Plants*, Safety Guide, Safety Standards Series, NS-G-2–10, IAEA, Vienna.

IAEA (2003b), *Maintaining the Design Integrity of Nuclear Installations Throughout their Operating Life*, INSAG-19, IAEA, Vienna.

IAEA (2004), *Software for Computer Based Systems Important to Safety in Nuclear Power Plants*, Safety Guide, Safety Standards Series, NS-G-1.1, IAEA, Vienna.

IAEA (2005a), *OSART Guidelines 2005 Edition*, Services Series 12, IAEA, Vienna.

IAEA (2005b), *Environmental and Source Monitoring for Purpose of Radiation Protection*, Safety Guide, Safety Standards Series, RS-G-1.8, IAEA, Vienna.

IAEA (2006a), *Fundamental Safety Principles*, Safety Standards Series no. SF-1, IAEA, Vienna.

IAEA (2006b), *A System for the Feedback of Experience from Events in Nuclear Installations*, Safety Guide, Safety Standards Series, NS-G-2.11, IAEA, Vienna.

IAEA (2006c), *The Management System for Facilities and Activities*, Safety Requirements, Safety Standards Series, GS-R-3, IAEA, Vienna.

IAEA (2006d), *Application of the Management System for Facilities and Activities*, Safety Guide, Safety Standards Series, GS-G-3.1, IAEA, Vienna.

IAEA (2006e), *Storage of Radioactive Waste*, Safety Guide, Safety Standards Series, WS-G-6.1, IAEA, Vienna.

IAEA (2007), *Arrangement for Preparedness for a Nuclear or Radiological Emergency*, Safety Guide, Safety Standards Series, GS-G-2.1, IAEA, Vienna.

IAEA (2008a), *Conduct of Operations at Nuclear Power Plants*, Safety Guide, Safety Standards Series, NS-G-2.14, IAEA, Vienna.

IAEA (2008b), *Improving the International System for Operating Experience Feedback*, INSAG-23, IAEA, Vienna.

IAEA (2009a), *Ageing Management for Nuclear Power Plants*, Safety Guide, Safety Standards Series, NS-G-2.12, IAEA, Vienna.

IAEA (2009b), *Evaluation of Seismic Safety for Existing Nuclear Installations*, Safety Guide, Safety Standards Series, NS-G-2.13, IAEA, Vienna.

IAEA (2009c), *Severe Accident Management Programmes for Nuclear Power Plants*, Safety Guide, Safety Standards Series, NS-G-2.15, IAEA, Vienna.

IAEA (2009d), *The Management System for Nuclear Installations*, Safety Guide, Safety Standards Series, GS-G-3.5, IAEA, Vienna.

IAEA (2010a), *Governmental, Legal and Regulatory Framework for Safety. General Safety Requirements Part 1*, Series no. GSR Part 1, IAEA, Vienna.

IAEA/NEA (2010b), *Nuclear Power Plant Operating Experience from the IAEA/NEA International Reporting System for Operating Experience 2005–2008*, IAEA, Vienna.

IAEA (2010c), *Deterministic Safety Analysis for Nuclear Power Plants*, Specific Safety Guide, Safety Standards Series, SSG-2, IAEA, Vienna.

IAEA (2010d), *Development and Application of Level 1 Probabilistic Safety Assessment for Nuclear Power Plants*, Specific Safety Guide, Safety Standards Series, SSG-3, IAEA, Vienna.

IAEA (2010e), *Development and Application of Level 1 Probabilistic Safety Assessment for Nuclear Power Plants*, Specific Safety Guide, Safety Standards Series, SSG-4, IAEA, Vienna.

IAEA (2011a), *Safety of Nuclear Power Plants: Commissioning and Operation*, Specific Safety Requirements, Safety Standards Series, SSR-2.2, IAEA, Vienna.

IAEA (2011b), *Chemistry Programme for Water Cooled Nuclear Power Plants*, Specific Safety Guide, Safety Standards Series, SSG-13, IAEA, Vienna.

24
Decommissioning of nuclear power plants (NPPs)

T. S. LAGUARDIA, LaGuardia & Associates, LLC, USA

Abstract: This chapter presents the historical basis for decommissioning experience, and summarizes the developments in cost estimating, planning, technologies, regulations, implementation, and waste management associated with nuclear power plant decommissioning. Examples of the application of this technology are included, and listings of international experience are provided. Sources of additional information are included for further investigation.

Key words: decommissioning planning, cost estimating, technologies, major component dismantling, waste management, international experience.

24.1 Introduction

Decommissioning of nuclear power plants (and fuel cycle facilities) has taken a long and sometimes tortuous path. The evolution of the decommissioning industry followed past practices from the demolition of non-radioactive fossil-fueled power plants and other process facilities which didn't present the difficulties of dealing with high levels of radiation and hazardous and toxic materials, nor waste disposal issues. The lessons learned from each new experience in decommissioning were passed on to later projects and slowly but constructively built a knowledge base on which to plan for the future decommissioning of newer, more complex units.

This chapter describes the historical experience of early decommissioning projects and how that knowledge provided the basis for responsive technical, financial, radiological safety and environmental planning throughout the world. This chapter describes the development of detailed and reliable cost and schedule estimating for decommissioning, and the need for standardization of costing methodology so cost comparisons can be made from one project to another and one country to another. The need for long-term planning is described to provide adequate funding for decommissioning so the work will be performed safely and efficiently. The techniques developed are also provided which permit planners to anticipate problems and circumvent them when segmenting reactor vessels and internals, dismantling piping and components, and demolishing structures.

The chapter includes a discussion of the details of each decommissioning phase to provide a framework upon which to consider the manpower and material resources needed to accomplish the work, and to understand the regulatory requirements to guide the planner in preparing for ultimate license termination. Since waste management is a major consideration to a successful decommissioning project, the chapter covers the essential issues that must be considered from the disposal facility waste acceptance criteria backwards to the waste packaging and transportation considerations. Similarly, waste recycling is a viable techniques to reduce the amount of waste that must be disposed of and accordingly some of the techniques for this technology are discussed.

The chapter also presents an overview of the international experience available as additional information sources to build upon the lessons learned from earlier projects. There are many international organizations dedicated to sharing technical and economic information for developing nations new to decommissioning, and numerous conferences, symposiums and workshops available to keep up to date with evolving technologies. Also included are suggested resources for additional information including government, institutional and commercial organizations producing regulations, handbooks and guideline documents to assist in the long-term planning for decommissioning.

24.2 Brief history of the development of decommissioning

Following the end of the cold war, the international community of major powers focused their attention to developing peaceful uses of nuclear energy. Many different technologies for power production were developed, each with a country-specific technology aimed at finding the most efficient and reliable source of energy to replace the fossil-fueled power industry. Governments invested huge amounts of financial resources, and companies dedicated their finest scientists to develop this newly found energy source. Several early reactor designs were constructed to test and learn the most effective methods for power production. Many of these early designs were fraught with problems not predictable in the design phase, and resulted in premature shutdown and decommissioning. The lessons learned from these early experiences provided valuable input to the next generation of reactors. Each major country developed its own technology, and worked out the problems to achieve remarkable gains in knowledge to provide workable solutions for many years of power production. The development of each reactor design required a coincident design to improve on fuel reliability, and ultimately to reprocess the fuel to recycle the reusable portion for future fuel fabrication.

The early efforts focused on production while issues related to waste management and disposal were deferred for later generations to resolve, as the belief was that if the technology to design new reactors were possible, so would be the ability to safely dispose of the wastes. This philosophy later came to haunt the industry designers as the socio-economic factors proved to be more challenging than anticipated. The costs of dealing with the wastes and ultimately with the final shutdown and decommissioning proved to be greater than expected.

At the same time, government regulation of these new technologies introduced challenges with respect to establishing standards of safety that could not be achieved without financially burdening the industry. The evolution of regulatory agencies and corresponding safety standards became a major hurdle for operating utilities to overcome. The very early designs could not be back-fitted to satisfy these new regulations, and the companies were forced to shut down the reactors and decommission them. The processes developed in decommissioning provided a knowledge base that was shared from one company to another and from one country to another.

However, the cost of decommissioning grew at rates not experienced in other developing industries, such as in the coal, oil, and gas industries, and companies were not financially prepared to handle these expenses. At the peak of the new nuclear design period in the 1970s and early 1980s, companies found the cost of construction growing to meet the regulatory standards imposed by regulators. The international economic situation tightened as fossil energy sources were strained by embargos on oil imports, and investors pulled back commitments to complete nuclear units under construction. Banks likewise tightened credit to companies investing in new nuclear plants, driving construction interest rates into double digits. A few companies faced potential bankruptcy, and some declared bankruptcy to avoid complete failure. Regulators realized the long-range liabilities of decommissioning were a significant risk, potentially leaving the cost of decommissioning as a future burden on shareholders, taxpayers, and rate-paying consumers.

State and federal regulators responded by forcing utilities to establish funding mechanisms to provide for these future liabilities. Many proposals were proffered to attempt to predict the future liabilities, and to regulate how the monies for ultimate decommissioning could be assured without depending on the individual utility's financial resources, long believed to be protected through the routine ratemaking process. The funding evolution process took decades to develop, and ultimately resulted in codified principles and practices that satisfied the assurance of funding availability.

As these early decommissioning programs evolved, the technologies used were constantly improving. They built on the experiences of previous programs and information was freely passed on to future projects. Experienced personnel brought their lessons learned to the next project, and built on the base of information technology to improve productivity, safety to the workers, the public and environmental protection. Government agencies funded additional research to test new techniques in small-scale demonstration programs, and published the results for contractors and vendors to use cost-effectively on large-scale projects. This technology transfer was international in scope, and organizations were formed to further promote new technologies for developing nations.

The most difficult tasks of segmenting large reactor components, such as reactor vessels and their internals, steam generators, and pressurizers, used creative adaptations of existing technology, and built on these technologies to improve productivity and safety. Advances in computerization and control systems led to the development of robotic systems of remotely operated arms to deliver cutting systems such as the plasma arc torch, high-pressure water abrasive jets, and mechanical cutting systems for the safe segmentation of these major components under water or in air.

New technologies in chemical decontamination of process systems were developed, and improvements in handling secondary wastes arising from decontamination were developed. Each new decommissioning project became a testing ground for new processes and the lessons learned were shared by means of reports, conferences, training programs, and technology transfer by experienced personnel moving from one project to another.

During this evolution of technologies, the methodologies for estimation of costs similarly advanced the science. Computer codes replaced hand calculations of cost estimates, eliminating routine calculational errors and improving the reliability and credibility of the estimates.

Cost feedback from actual decommissioning projects and operating plant modifications provided a database for estimating future costs with greater reliability. More sophisticated computer codes for scheduling provided the ability to estimate complex multiple dismantling activities, and the concepts of critical path analyses provided a means to improve productivity from the workforce.

Decommissioning financing models were developed and regulatory agencies imposed requirements to establish external trust funds outside utility control to assure that adequate funding would be available when decommissioning was implemented. State regulatory agencies quickly adopted these new funding schemes as a means to avoid the potential risk to ratepayers of having to supplement inadequate funds. International regulatory agencies similarly adopted these funding mechanisms and provided guidelines for implementation within developing nations. No other industry had previously adopted this approach. It became a model for regulation of ore mining companies, oil companies, and coal companies to use for future disposition of closed mines and oil wells.

Through the more recent deregulation process, nuclear utility companies established merchant companies that were unregulated with respect to the rates to be charged. Some of these companies recognized the long-term value of nuclear power plants and initiated acquisition programs to buy faltering power plants, or single-unit plants that could not compete in a competitive market. The cost of decommissioning became one of the major considerations in the purchase price, and the transfer of existing funds to the new owner an issue of negotiations. The terms of sale included how the funds would be transferred, and how funding adequacy would be assured in the future.

24.2.1 Development of peaceful uses of atomic energy

Following the end of World War II, those countries leading in the technology began to focus their attention on the peaceful uses of nuclear energy. US President Dwight David Eisenhower made his famous speech to turn swords into plowshares, and the nuclear power industry took off with one of the greatest technological advances known to man. The early efforts to build nuclear weapons by all the major companies such as DuPont, Union Carbide, W.R. Grace, Allis Chalmers, Ashland Oil, and many others now turned to developing nuclear power reactors. The federal government provided generous funding, and some of the finest minds in the industry were recruited to advance this technology. Other countries such as Canada, the UK, France and the former USSR took up the challenge to develop various reactor types to determine the best technology for the safest use of this new fuel. Germany started later when the prohibition on developing nuclear energy was lifted.

The industry was faced with the challenge not only to design, construct, and operate these new types of plants, but also to deal with the rest of the fuel cycle to determine how to make the fuel, transport it safely, remove spent fuel, and safely dispose of the fuel and its wastes. Most countries developed reprocessing plants to extract the still reusable portion of the spent fuel and plutonium, and to separate out the wastes to be stored in tanks until a permanent disposal technology was available.

Many of these early reactors and fuel cycle facilities designed in the late 1950s were small demonstration reactors, designed to test a specific technology and prove the economics of nuclear power. The federal governments of all these countries funded most if not all of the early development to encourage private industry to step in and take the lead on future plants. Some demonstration reactor types quickly proved to be impractical, and were shut down for decommissioning. Reactors like Hallam in Nebraska, Piqua in Ohio, and Elk River in Minnesota operated for only a few years before developing significant irreparable piping leaks or other fuel-related problems. In Europe, the Organic Cooled Reactor attracted a great deal of interest, and the 20 MWth high-temperature gas-cooled DRAGON research reactor project at Winfrith, UK, was a cooperative activity within the OECD Nuclear Energy Agency. The BONUS reactor in Puerto Rico was designed to generate high-pressure steam to use a conventional, more efficient steam turbine. The reactor vessel contained both boiler fuel to boil the water and superheater fuel to increase its temperature. The dual-fuel power level proved too difficult to control and the reactor was shut down for decommissioning. Further, the plant was designed with the control room and all management offices under the same containment dome as the reactor, so if there ever were a serious accident, serious injury would be most likely. This early experience proved valuable in sorting out the successful reactor designs and set the stage for larger, more efficient reactors to be built in the future.

24.2.2 Early reactor designs and operating experience

By the early 1960s and into the 1970s, reactor designs improved substantially and fuel designs became more reliable. Each country seemed to adopt a specific reactor type, and reactor vendors competed aggressively to supply utilities with their designs. In the US, two designs predominated, namely the pressurized water reactor (PWR) and the boiling water reactor (BWR) using low enriched fuel (3–4% enrichment in U-235). Both were extrapolations of US Navy submarine and aircraft carrier nuclear power plants, with a solid proven operating history. For its base design, Canada selected a heavy water-moderated, pressure-tube reactor design that used naturally enriched fuel. This type would later be called the CANDU reactor. The UK adopted graphite-moderated gas-cooled reactors and used naturally enriched fuel. France used both the graphite-moderated gas-cooled reactors and the sodium-cooled reactors for their higher thermal efficiency, though later on, in the early 1980s, France abandoned the gas-cooled reactors for the PWRs. The Soviet Union also adopted the graphite-moderated boiling water cooled reactor, RBMK, for electricity production and also the PWR type. The RBMK was the most relevant graphite-moderated reactor developed by the USSR for electricity generation.

When the mid-1970s brought the world oil embargo, nuclear power seemed the easy answer to reduce our dependency on oil. Utility companies were impressed with this new technology and began ordering multiple units envisioning 'energy parks' with as many as 10 reactors on a site. Government support was strong and the future growth potential seemed unlimited. All countries seemed to be joining the movement to increase the use of nuclear power.

24.2.3 Early decommissioning regulation

During the mid-1970s, government regulatory agencies were coming under severe criticism by the anti-nuclear interveners and the public, accusing the government of being both the promoter of nuclear power and the regulator of its safety. This was an apparent conflict of interest and pressure was brought to bear to separate these functions. In the US in 1975, the old Atomic Energy Commission (AEC) was changed to the Energy Research and Development Administration. The government then transferred the design safety responsibility to a new agency called the Nuclear Regulatory Commission (NRC). The pattern established in the US in creating the independent NRC was closely followed by other countries. For instance, in 1980 the Spanish government created the independent Nuclear Safety Commission and more recently France created the Authority on Nuclear Safety, both institutions similar to the US NRC. Actually, most nuclear countries have similar institutions.

This new agency began to focus its attention on the design safety of nuclear plants, and to draft new regulations to address issues formerly treated as routine concerns as for a fossil plant. The recognition of a possible accident resulting from an earthquake generated new regulations for seismic design. The NRC funded major accident studies, resulting in the redesign of systems to meet accidents caused by a loss of coolant. The high-energy systems in a power reactor raised concerns over the consequential damages from a pipe rupture causing a pipe to whip back and forth, rapidly taking out other safety systems in its path. Pipe whip restraints were designed to handle these enormous structural loads, and then were back-fitted into existing (already tight) spaces in operating reactors. The potential for a common mode failure was examined, where a single event (explosion or airplane impact) could take out critical safety systems, and drove the NRC to require a new design concept. Plants would have to be redesigned for redundancy (more than one power source – two pumps to perform the same safety function), diversity (two different types of power – such as electric and diesel driven), and separation (one pump at one side of a building and the other on the opposite side). All these design features had to be back-fitted into the existing plants and new plants had to incorporate them to obtain their construction license.

During this same time, the US economy fell on hard times and banks tightened up on credit, especially to the uncertainties of nuclear power plants. Wall Street steered investors away from investing in any utility with nuclear plants. Interest on construction loans grew to as much as 20–25% – unheard of in any previous situation. More than 60% of the cost of constructing a new nuclear plant was for interest on construction loans.

Suddenly, the cost of new nuclear plants rose by factors of 10 or more. The commercial nuclear utilities felt the impact of these enormous design and construction costs, and nuclear power fell out of favor. Some utilities tried to absorb these large increases in cost and had to declare bankruptcy. The NRC responded by requiring financial assurances that a utility could complete decommissioning of any plant contemplated or currently under construction.

To complicate the problem further, a serious accident occurred at the Three Mile Island Unit 2 nuclear plant in 1979. Virtually the entire core was damaged, though only about 20% of it melted and no harmful radiation was released to the environment. Every regulatory agency in the US became involved in trying to understand the causes and implications of the accident, and new regulations were formed, requiring that utilities further back-fit operating plants to record all on-going events in case of an accident. The TMI-2 accident was closely analyzed by every nuclear regulator in the world and within the IAEA and the NEA/OECD where a common research effort was established to find the causes and consequences of the accident. Many millions of dollars were spent to upgrade these plants, and nuclear power again fell in greater disfavor around the world. Some countries such as Italy, Sweden, Germany, and others decided to shut down all their reactors, even though they provided more than half their electricity generation. The TMI-2 accident had a great influence on nuclear development in Europe, but the effective moratoria in Italy and the promised but never fully achieved moratoria in Sweden were initiated by the 1986 Chernobyl-4 accident. The limitations in the operating lives in the German plants came much later. Decommissioning became the catchword of the future.

24.2.4 Early decommissioning experience

The early demonstration reactors such as Hallam, Piqua, Elk River, and BONUS that were shut down provided a training ground for the development of decommissioning technology. The concepts of entombment, safe storage, and complete dismantling were developed, and the AEC used these early experiences to formulate regulatory guidance for future decommissioning programs. The NRC Regulatory Guide 1.86, *Termination of Operating Licenses for Nuclear Reactors*, provided specific guidance on each potential decommissioning strategy (US AEC, 1968). Although primitive in its form and content, it provided initial direction for the reactor owners to decommission these early reactors safely. Hallam, Piqua, and BONUS used the entombment strategy for decommissioning, removing the spent fuel and entombing some or all of the remaining radioactivity within the containment structure in a structurally sound concrete barrier. In 1967, Hallam was buried within a mound of soil and monitored periodically. In

1967, Piqua removed all equipment except the Reactor Vessel (RV); it was filled with gravel (for shielding) and the building was temporarily used for storage of industrial equipment. In 1969, BONUS systems were decontaminated using strong acids, and the RV and empty fuel storage pool were entombed within a concrete barrier. BONUS was to be used as a museum for the public to see what a reactor looked like; it remained open for about two years. In 1973, the Elk River reactor was completely dismantled and all radioactive components were shipped to federal disposal facilities.

The Shippingport Atomic Power Station, the first commercially operated power plant in the US, was shut down in 1982, and the reactor components and buildings were completely dismantled. Except for the turbine-generator systems, all vestiges of the facility were removed and the site returned to a 'greenfield' status (IAEA, 1989; Tarcza, 1987).

The US applied this early guidance to industrial reactors and a number of research reactors installed at universities, research laboratories, and medical facilities. Other countries followed suit with similar requirements and safely decommissioned research reactors and some of the early experimental reactors.

The Canadian prototype power reactors Gentilly Unit 1 and Douglas Point were developed to demonstrate the CANDU technology. Both were shut down for decommissioning: the Gentilly Unit 1 250 MWe pressurized heavy water reactor in 1977; the Douglas Point 200 MWe CANDU reactor in 1984. The NRX, the experimental materials reactor at Chalk River, was shut down for decommissioning in 1993.

24.2.5 Technological developments in decommissioning

As noted earlier, these early decommissioning experiences led to the development of new technologies. At BONUS, chemical decontamination was first used on a large scale to remove the internal contamination from piping and components to make them safe for public access. The reactor vessel was entombed in a concrete structure designed to last 125 years to allow the residual radioactivity to decay to unrestricted access levels. More than 500 cubic yards of concrete were pumped under, around, and on top of the RV. Because of the high ambient temperatures in Puerto Rico, and because the heat of hydration of concrete during curing would cause cracking in the concrete, 100 pounds of ice had to be added to each cubic yard of concrete mix before it could be pumped the 285 feet into the building.

At Elk River, the RV internals were segmented remotely under water using a remotely operated plasma arc torch developed by the Oak Ridge National Laboratory in Tennessee. The vessel was similarly cut in air with torch cutting. These technologies were subsequently used at the Sodium Reactor Experiment in Santa Susana, CA, and more recently at the Yankee

Rowe plant in Massachusetts. At Shippingport, concrete scabbling (scarification) was used to remove contaminated concrete from the spent fuel pool walls and floors; this large-scale project used a multi-head scabbling machine designed to scarify two fuel pool walls simultaneously. The concrete containment structure at Shippingport was demolished using large hydraulic hammers mounted on backhoes and excavator machines. These technologies are still in use today.

24.3 Development of decommissioning cost-estimating methodologies

The growth of this new technology for nuclear power did not emphasize the costs for decommissioning. Little thought was given to the ultimate cost to dismantle and then dispose of the wastes. As the industry grew, regulators, industry groups, concerned citizens and other stakeholders began to investigate decommissioning costs. The early estimating tools from the construction industry did not properly address the radiological decontamination, removal, and ultimate disposition of these facilities. The cost estimating community of engineers embarked on an intensive effort to address these costs. The work set a benchmark for other industries to follow as to how to deal with these long-term liabilities. This section describes the development of the estimating methodology and its application to decommissioning.

24.3.1 Planning for future decommissioning – cost estimating and financing

During the development years of nuclear power from the 1960s through the 1970s, the focus was to build new and larger nuclear power plants. Little concern was given to ultimate decommissioning. The costs of decommissioning were believed to be a small fraction of the cost of building the plants – less than 10%. In fact, many estimates stated that figure without further substantiation. If the industry could build these plants, they could find a way to decommission them inexpensively.

During these early years, the cost of radioactive waste disposal was truly inexpensive. In the US, low-level wastes were being buried at shallow land burial facilities for as little as $0.75 per cubic foot. Higher-activity wastes carried a surcharge for radiation level and total curies amounting to only a few dollars per cubic foot. Waste disposal facilities essentially dug a trench about 27 feet deep, placed the waste in the trenches, and covered it with soil. Regulations for decommissioning were on minimal, and labor and energy costs were on about the same level as the Consumer Price Index.

Some of these disposal facilities experienced leakage of radionuclides from the trenches. In the US, at Morehead, KY, Sheffield, IL, and Barnwell, SC, traces of waste were detected leaking off site, and major changes were made. Morehead and Sheffield were closed. Barnwell had to recover these leakages, and then had to redesign the trenches. A clay bottom was installed, and a French drain system of piping was installed on top of the clay, and then covered by gravel and another clay layer. The walls of the trenches had to be sealed with clay to prevent side leakage. The imposed state and federal regulations and subsequent inspections ensured the integrity of disposal. The cost of disposal of wastes increased from \$0.75 per cubic foot (\$0.02 per m^3) to as much as \$75.00 per cubic foot (\$2.12 per m^3), a 100-fold increase. The type of containers that could be used for disposal now had to be strong-tight containers of steel. Higher-level wastes such as contaminated resins required polypropylene containers reinforced with steel structures to prevent subsidence of the overburden of soil.

The early estimates of 10% of the construction cost to pay for decommissioning were no longer applicable. State and federal regulators became concerned that funding adequacy was not available. By 1986, the NRC enacted new regulations requiring licensees to show evidence that at least \$100 million would be available to pay for decommissioning. Commercial utilities attempted to estimate decommissioning costs using out-dated methodologies that fell far short of the goal. Inconsistencies in estimating were rampant and public and regulatory confidence was lost.

Various independent cost-estimating companies used a variety of approaches to estimate decommissioning costs, but the results differed so greatly that there was no confidence in the estimates or whether the nuclear industry could properly predict future costs.

24.3.2 Recognition of the need for accurate cost estimates

As noted earlier, original estimates used a value of 10% of the construction cost to estimate decommissioning costs. This was partly based on fossil-fueled power plant demolition experience, where clean, non-radioactive materials were sold as scrap to recover some of the dismantling costs. But even this practice was flawed, since demolition companies often removed the more valuable copper and steel from the fossil-fuel plants and abandoned the project partially dismantled, leaving the owners with the problem of removing asbestos insulation and concrete structures to complete the job.

State regulators who set the rates for electricity for the commercial nuclear utilities were frustrated with the wide diversity of estimating methods and lack of consistency in the estimates from one plant to another.

Interveners tarnished the image of utilities in public hearings, accusing them of artificially inflating the cost of decommissioning to increase the rates to be charged to rate-paying consumers. In 1978, the NRC contracted an independent national laboratory, Battelle Pacific Northwest Laboratory, to develop a reference study of decommissioning large PWRs, and later BWRs and gas-cooled reactors (and other fuel cycle facilities) (Smith *et al*., 1978; Oak *et al*., 1980). The NRC hoped to use these studies to establish a benchmark against which estimates could be developed and compared across the US.

The Atomic Industrial Forum (now the Nuclear Energy Institute – NEI) funded a study to develop a cost-estimating methodology to estimate costs accurately. The report, *Guidelines for Producing Nuclear Power Plant Decommissioning Cost Estimates*, was published in 1986 (hereinafter, 'the Guidelines'), and set the groundwork for a new approach to estimating (LaGuardia *et al.*, 1986). The AIF retained a peer group to review and direct the study to best serve the industry. With the goal of instructing how to develop an estimate without directly publishing a specific estimate, the Guidelines provided a step-by-step methodology for estimating and provided extensive bases to support the calculations.

Utilities contracted with independent companies (for example, TLG Services, Inc.) to develop estimates with the Guidelines for decommissioning their facilities. The first Guideline-based estimates proved to be higher than the Battelle reference plant estimates, and state regulators were suspicious that again the utilities were attempting to inflate costs to increase their rates. Interveners pressed expert witnesses to justify the differences between the Guidelines and the Battelle studies, and a battle-royal ensued. Upon further examination, witnesses showed that the Battelle studies contained some flaws in both estimating assumptions and the inventories of equipment. Some of the data used by Battelle were from plants that were still under construction and not all equipment was accounted for properly. The NRC held meetings with Battelle and TLG Services to reconcile these differences; some progress was made, but Battelle (and the NRC) refused to back down. Nevertheless, the Guidance methodology prevailed in the state hearings and state regulators agreed to the amounts required to decommission these large plants.

Other countries, including Canada and Europe, watched the development of this cost-estimating technology with great interest. In Europe, the German company NIS Ingenieure (now part of Siemplekamp, Inc.) began to develop estimates for utilities using the decommissioning cost results of a small heavy water-moderated, gas-cooled reactor as a basis for estimating all types of power reactor decommissioning costs. The NIS estimates were a 'black box' approach, where the details of the estimate were not revealed to the utility. Nevertheless, utilities relied on these estimates.

24.3.3 Funding for decommissioning

Construction costs grew considerably (by almost a factor of 10) during the early 1980s – the peak years of nuclear plant construction. The increase was partly due to the high cost of interest, and partly due to the new and continually changing regulations imposed by the NRC. In the US, New Hampshire Yankee, owners of the Seabrook plant under construction, declared bankruptcy, sending panic waves throughout the industry. Utilities never declared bankruptcy as they were supposed to be protected from such by the regulating rates of public service commissions. Following the Three Mile Island Unit 2 accident, other companies threatened bankruptcy at having to back-fit their plants with the modifications needed to protect the plants during an accident.

The NRC followed with new regulations requiring licensees to set aside funds for decommissioning to ensure the availability of funds when needed. These new regulations took several years to develop because the NRC changed form and content several times in an attempt to find a reasonable approach to guarantee assurance.

24.3.4 Attempts to establish minimum funding amounts by formula

The NRC's approach for funding assurance was first to establish a minimum funding amount of $100 million for each reactor unit. It cited the Battelle studies as a basis, and felt this simplified approach would be satisfactory for ratemaking purposes and funding adequacy. However, with the rapidly increasing costs of decommissioning from waste disposal charges, new regulations, and better-defined decommissioning plans, the $100 million was viewed by state regulators as inadequate. The $100 million was for 1000 MWe power plants and did not address smaller or larger plants. State regulators placed greater reliance on the site-specific estimates developed using the Guidelines methodology. Now the regulators were insistent that sufficient funds be set aside to ensure the state would not be left responsible for decommissioning the plants in case of utility bankruptcy. The tide had turned in favor of greater assurance.

The NRC responded by developing a formula incorporating a base cost for each type of reactor (PWR and BWR) of a given size, and a modifier to take into account plant size, labor costs, disposal costs, and energy costs. Regional differences were adjusted using consumer price indexes according to specific references directed by the NRC'. These guidelines were published in the US NRC's *Standard Review Plan for Decommissioning Cost Estimates for Nuclear Power Reactors* (US NRC, 2004). Licensees of operating reactors were now required to provide reasonable assurance that funds would be available to accomplish decommissioning within 60 years from the date of permanent cessation of operations. These requirements ensure that a licensee has financial assurance in effect for an amount that may be more but not less than the minimum funding amount (MFA). Accordingly, the NRC's formula stated that if P equals the thermal power of a reactor in megawatts (MWt), the MFA (in millions, January 1986 dollars) is:

$$\text{For a pressurized water reactor (PWR): MFA} = 75 + 0.0088P \qquad [24.1]$$

$$\text{For a boiling water reactor a (BWR): MFA} = 104 + 0.009P \qquad [24.2]$$

For either a PWR or a BWR, if the thermal power of the reactor is less than 1200 MWt, the value of P to be used in these equations is 1200, whereas if the thermal power is greater than 3400 MWt, a value of 3400 is used for P. That is, P is never less than 1200 or greater than 3400.

The financial assurance amounts calculated in the above equations are based on January 1986 dollars. To account for inflation from 1986 to the current year, these amounts must be adjusted annually by multiplying by an escalation factor (ESC). This ESC is:

$$\text{ESC (current year)} = 0.65L + 0.13E + 0.22B \qquad [24.3]$$

where L and E are the ESCs from 1986 to the current year for labor and energy, respectively, taken from regional data of the US Department of Labor (US Department of Labor, 2010b), and B is an annual ESC from 1986 to the current year for waste burial, taken from the most recent revision of *Report on Waste Disposal Charges: Changes in Decommissioning Waste Disposal Costs at Low-Level Waste Burial Facilities* (US NRC, 2005). This document is updated from time to time to account for disposal charge changes. In January 1986 (the base year), using disposal costs from the DOE's Hanford Reservation waste disposal site, L, E, and B all equaled unity; thus, the ESC itself equaled unity. Therefore, the minimum funding amount (MFA) is:

$$\text{MFA (in millions, current year dollars)} = \text{MFA} * \text{ESC (current year)} \qquad [24.4]$$

For example, a 2536 MWth BWR decommissioning cost in 1986 for immediate dismantling would have been \$126.82 million, and in 2002 with the escalation factors applied would be \$424.36 million. The escalation factor calculation is as follows:

$$\begin{aligned}\text{ESC} &= 0.65L + 0.13E + 0.22B \\ &= (0.65 * 1.922) + (0.13 * 1.135) + (0.22 * 8.86) = 3.346 \qquad [24.5]\end{aligned}$$

The coefficients in the US NRC formula were taken from the Battelle studies. They represent the percentage of the total costs that were related to labor (65%), energy (13%), and burial (22%). Accordingly, this escalation formula only applies to the NRC reference plants, not to a site-specific study that would have different coefficients.

For estimating purposes, the US NRC formulas were used as benchmarks for determining the minimum funding amount. However, most utilities and state regulators preferred to rely on site-specific studies.

24.3.5 Advent of site-specific cost estimates

The lack of specificity in the US NRC formulas drove utilities and state regulators to require site-specific studies for funding purposes. Site-specific studies by their very nature more closely identified the actual reactor and site conditions that accounted for the cost of decommissioning a nuclear power plant. The site-specific studies required a detailed itemization of all the equipment and structures at a facility, and the exact end-point conditions for decommissioning. The Guidelines methodology depended on a 'building block' approach to cost estimating. Estimates were developed for activity-dependent costs (decontamination, removal, packaging, transportation, and burial), period-dependent costs (management cost for the utility and decommissioning operations contractor for the duration of the project), and collateral costs (engineering, purchased equipment, licensing/permitting/insurance, and taxes). The total costs were then analyzed to add contingency (to account for costs fully expected to be incurred during actual decommissioning, but whose occurrence, duration, and degree of uncertainty could not be directly calculated).

To determine the activity-dependent costs, the plant inventory of equipment and structures was used as input to the calculations. Unit cost factors (costs per foot of pipe of each size, cost per ton of steel, and cost per cubic yard of concrete) were developed using the local site labor costs and materials costs. For example, the number of feet of a specific pipe size was multiplied by the unit cost to remove that pipe and bring it to a packaging staging area for ultimate disposition. Period-dependent costs were developed from the project schedule of the overall program, and the management staff was adjusted for the various periods of the project from preparations to operations (removal of radioactivity), and finally site restoration (dismantling of all remaining structures and restoration of the site). Collateral costs included all costs that were neither activity nor period-dependent.

24.3.6 Development of defined funding programs – internally held vs external trusts

The utilities used these site-specific studies to develop funding programs to ensure the funds would be available when decommissioning commenced.

Some utilities initially adopted internally held funds, but the uncertainties associated with potential bankruptcy drove the US NRC to require funds to be externally held by trusts (banks or certificates of deposit) so that utilities could not use the funds for other purposes. The US NRC established guidelines for the type of external funds that could be used.

The US NRC originally required safe-type investments similar to the funds established for coal miners potentially suffering from 'black lung disease'. They were in low-return but secure investments to ensure availability when needed. The Internal Revenue Service (IRS) referred to these funds as qualified funds and allowed a lower tax rate on the utilities. Later, the US NRC and the IRS allowed investments in non-qualified funds (equities and stocks) that provided higher returns but were potentially at a higher risk, and were taxed at the full corporate tax rate. The overall greater return proved to be less costly to consumers, but still provided adequate assurance that the funds would be available when needed.

When deregulation of nuclear utilities was promoted in the US with the intention to reduce the cost of electricity to consumers, utilities formed unregulated 'merchant companies' and were permitted to retain these collected funds provided the utilities would guarantee the assurance of all decommissioning costs. Most utilities retained the external trust concept, but were able to earn interest on the funds as profits while committing to have the funds available for decommissioning. Any shortfall would have to be made up from the shareholders of the utility.

As actual decommissioning experience was learned, the costs of decommissioning increased. Changes in the scope of decommissioning were recognized to comply with the increasing regulatory requirements and the increasing effects of inflation and waste disposal costs. As noted earlier, waste disposal costs were increasing at rates greater than the Consumer Price Index and were driving decommissioning costs higher. The concern for the environment and the disposition of materials drove costs higher. Local stakeholders gained greater control as to the final disposition of the site, and forced utilities to expend more dollars to restore the site to lower residual levels of radioactivity at greater cost.

In light of these changing conditions in scope and costs, state and federal agencies recognized the need for periodic updating of cost estimates. Public service commissions that regulate rates charged to consumers built periodic updating into the funding programs. Most state regulators require a funding review every three to five years. These reviews are held at hearings where the utilities present the latest updated costs and calculate what the revised liability is, and how much should be collected from rate-paying consumers.

The US NRC similarly recognized the need for updating, and in 1999 issued regulations requiring licensees to submit estimated decommissioning costs and funding plans to demonstrate there would be adequate funds when needed. Every two years, utilities file an updated estimate to show that funding meets or exceeds the NRC's minimum funding amount (MFA). Licensees that are in a shortfall condition are notified to make up the shortfall, or be subject to an NRC fine.

24.4 Development of long-term planning for decommissioning

The development of long-term planning includes preparation of decommissioning plans in accordance with regulatory requirements, anticipating the technologies required to decontaminate systems, dismantle and demolish equipment and structures, transport materials for storage/disposal, and recycle or dispose of wastes. This section provides an overview of this long-term planning.

24.4.1 Requirement for decommissioning plans

The early decommissioning programs for small demonstration and research reactors required minimal documentation for planning and approval. There was no precedent for licensees/owners and regulators to follow. Accordingly, licensees/owners would propose an approach and the regulators would review it. Numerous regulator questions arose and multiple submittals were necessary to arrive at a consensus of safe practices and acceptance criteria. The process was tedious to say the least, and often frustrating and time consuming. Meanwhile, no physical work could be performed at the site other than those activities that were previously approved under the operating license. For this work, licensees/owners merely had to perform an internal review to assure that there were no unreviewed safety issues.

In the US, the former Atomic Energy Commission (US AEC) had begun to use a series of Regulatory Guides for operating reactors on specific activities for PWRs and BWRs. These numbered Regulatory Guides provided licensees/owners with pre-approved approaches to safely operate and maintain the reactors without being subjected to intensive and time-consuming submittals and approvals. With respect to decommissioning, the US AEC prepared and issued Regulatory Guide 1.86, *Termination of Operating Licenses for Nuclear Reactors*, in June 1968 (US AEC, 1968). This document was the first recorded attempt to codify rules for decommissioning. It required that a decommissioning plan be prepared and submitted for US AEC approval prior to the commencement of any fieldwork, and provided guidance on levels of radioactivity acceptable to free-release a site for unrestricted use and terminate the license. Regulatory Guide 1.86 also identified three decommissioning scenarios (strategies) acceptable to the

US AEC, including mothballing (safe storage), entombment (hardened safe storage), and dismantling.

The significance of this Regulatory Guide was important, as it gave licensees/owners direct guidance how to decommission a reactor with minimal regulatory submissions. Most of the early demonstration and research and development reactors were decommissioned in accordance with this document. Internationally, other countries followed this development cautiously, preferring to continue the previous practice of submittal, review, and re-submittal.

As experience was gained from actual decommissioning, new issues were raised involving variations in the scenario or strategy to be employed. As noted earlier, the US AEC was reconfigured into the US Nuclear Regulatory Commission (US NRC) and the importance of decommissioning funding and technical approaches drove an entire reassessment of guidance and requirements. Other countries similarly recognized the changes taking place in the industry and began to develop regulations and guidance for decommissioning. The International Atomic Energy Agency (IAEA) through consultant contributions and studies developed guidance practices and safety documents for its member states for nuclear power plants, research reactors and fuel cycle facilities.

24.4.2 Regulations for decommissioning

It is not possible to fully describe the regulations of each country as there are too many variations and differences. Section 24.5 provides additional sources of information on decommissioning regulations in selected countries. Many countries follow the guidance of the International Atomic Energy Agency which is sufficiently representative of common practice. For example, the IAEA issued the Safety Guide, *Decommissioning of Nuclear Power Plants and Research reactors* (IAEA, 1999), the Safety Requirement, *Decommissioning of Facilities Using Radioactive Materials* (IAEA, 2006a), and *Decommissioning of Nuclear Facilities: Training and Human Resources*, Nuclear Energy Series (IAEA, 2008).

24.5 Decommissioning technologies and research and development

International decommissioning experience has driven the need for new and more effective technologies to improve safety and productivity, while reducing time and cost. Necessity is the mother of invention, and this is no better demonstrated than in the decommissioning industry. All the creative juices of engineers and contractors were aptly applied to simplify the work and make it safer to accelerate programs without endangering the workers or public. Several of these areas are described in the following sections.

24.5.1 New technologies for *in-situ* chemical decontamination

Liquid-cooled nuclear power plants operate under strict chemical controls to maintain demineralized water purity to minimize corrosion of the critical components such as the reactor coolant pumps, steam generators, pressurizers, piping, reactor vessel, and the fuel. Nevertheless, slow corrosion and erosion continue to occur, causing fine particles to flow through the system and pass over the fuel. These particles become highly irradiated and collect on the interior walls of various components of the coolant system. Over time, these particles increase the dose rate to workers performing maintenance on the equipment. During decommissioning, the dose rates are likely to be at their highest levels if not removed periodically. Many components cannot be accessed for dismantling until these exposure rates are reduced or removed prior to dismantling. The nuclear industry has spent many millions of dollars to find technologies to remove this contamination safely.

In the 1970s, Canadian companies successfully developed the proprietary Candecon and Canderem decontamination methods to remove or reduce contamination in the CANDU reactors. They involved a multi-step process of injecting chemicals into the system and circulating them for several days while monitoring the decontamination factor (DF) achieved. When no further reductions were possible, the system was deemed decontaminated and safe for dismantling. This same technology was successfully applied in the US for several years.

Subsequently, US companies developed similar multi-step processes including the AP-Citrox/EDTA process (alkaline permanganate, followed by citric acid, followed by ethylene-diamine-tetraacetic acid). The chemical reactions are far beyond the purpose of this book, but suffice it to say they were very successful. This process was used successfully at the Shippingport Atomic Power Station in Pennsylvania. At the Dresden reactor in Illinois, a proprietary process called NS-1 (Nuclear Solvent-1) was used successfully to reduce exposures.

EPRI promoted and funded the 'Decontamination for Decommissioning' (DFD) process employing difluoroboric acid, a process developed by Bradtec in the UK. This process was used successfully at the Big Rock Point Decommissioning Project in Michigan.

Siemens, a German company with headquarters in Erlangen, Germany, developed the CORD process (Component Oxidation–Reduction Decontamination), employing a multi-step chemical addition. This process offered simplified end-point solvent removal using a photo (light-reducing) method. This process was successfully used at the Connecticut Yankee (Haddam Neck) Decommissioning Project in Connecticut.

In Germany, electropolishing was developed to decontaminate and free-release components from the Gundremmingen-A nuclear power plant.

Electropolishing is reverse electroplating, where an acid bath is used and a direct current source is applied to transfer electrons from the contaminated surface to the acid and the contamination is removed with the base metal. Variations of this technology have been developed whereby the acid is sprayed on the surface from a nozzle that is electrically charged.

As can be observed, the technology for decontamination is not stagnant. New advances are being developed as the industry gains experience. The successes of each of these projects are widely distributed through conferences, courses, and training programs.

24.5.2 New technologies for pipe and component removal

Fossil plant dismantling has made use of oxyacetylene (oxygen and acetylene gas) torches to cut steel components successfully for many years. These were manually held torches and were adequate for the purpose intended.

When nuclear decommissioning evolved, the oxyacetylene could not cut through the stainless steel components since it did not have a high enough temperature (1150°C) to melt the stainless steel. The plasma arc torch was introduced, using argon gas to start, then oxygen gas to cut. The torch was energized with a voltage across the tip to the workpiece to maintain the plasma heat source. The temperature of the torch tip was about 7800°C, and it cut through stainless steel without a problem. It cut through plain carbon steel about 10 times faster than oxyacetylene torches. This technology was adopted at the Shippingport Project as the main cutting tool for all cutting.

Plasma and oxyacetylene torches are expensive to operate, so the gasoline torch was developed. It used ordinary gasoline as the fuel and oxygen for cutting. It burns at about 4475°C, more than sufficient to cut through stainless steel and, of course, carbon steel. It is quickly becoming the standard tool for such work.

Laser cutting has been available for many years, but the size of the power supply and the limitations on the thickness that can be cut are not yet sufficient to be of great service at this time. Additional development work will improve this tool as well.

24.5.3 Reactor vessel and internals segmentation

The technology for segmenting the reactor vessel and its internals has been evolving steadily since the first attempts at the Elk River Reactor in Minnesota. The Oak Ridge National Laboratory in Tennessee adapted the plasma arc torch to a remotely manipulated central mast to deliver the torch tip to the workface. The torch worked extremely well for segmenting the stainless steel internals. The only problem encountered was maintaining the clarity of the water in the vessel needed to provide shielding. The filtration system was inadequately sized, and the years of 'crud' (Chalk River Undefined Deposit – a Canadian term coined for this material) buildup clouded the water such that the job had to be shut down periodically for the filters to catch up. This same problem has been encountered in every other process used for vessel internal segmentation in the US. The RV was segmented using an oxyacetylene torch on the same mast since most of the vessel was carbon steel. The Elk River torch was rebuilt and used again at the Sodium Reactor Experiment in Santa Susana, CA.

The next opportunity for segmenting reactor vessel internals was not until the 1990s at the Yankee Rowe Reactor, in Rowe, MA. Once again, the plasma arc torch was used, mounted on a bridge crane on top of the RV. The torch cut the stainless steel internals without a problem, but again the water clarity was a problem. In addition, because the reactor had operated for more years than Elk River, the fine particles created by the cutting torch were carried upwards by the thermal currents generated by the high heat input to the water. These particles caused an exposure rate problem for workers on the bridge, and additional shielding had to be added to protect them during cutting.

By the late 1990s, the Connecticut Yankee nuclear plant was the next in line to be decommissioned. For these RV internals, a new technology was developed. An abrasive water jet cutting system was employed. This consisted of a high-pressure water jet (2760 bars) with abrasive-injection at the cutting tip. The cutting rates were slower than for plasma arc cutting, but it could cut thicker sections not possible with the plasma torch. However, the large amount of abrasive (grit) needed (about three pounds of grit for each five gallons per minute of water, since the jet required about seven gallons per minute) created a large amount of secondary waste. This waste, further contaminated with the fine particulate from the internals it was cutting, required special containers for disposal. In addition, the grit spray was difficult to control after leaving the kerf (the slit cut by the torch) and spread this material all over the bottom of the pool where the internals were being cut. It took another year to clean up this debris from the pool.

Early in 2000, Maine Yankee initiated segmentation of the RV internals (Fig. 24.1). They too used the abrasive water jet technology. Once again, water clarity became a problem and a specially designed filter system was employed to replace the original system. The internals were successfully segmented, the more highly radioactive portions were stored with the spent fuel for ultimate disposal, and the lower-level components were transported to Barnwell, SC, for disposal.

In February 2001, the San Onofre Unit-1 (SONGS-1) nuclear plant in San Clemente, CA, initiated segmentation of the RV internals using the abrasive water jet technology (Fig. 24.2). They encountered the same water

24.1 Maine Yankee segmented reactor vessel internals.

clarity problems as had previous companies, and had to provide a specially designed filtration system. The work was completed in January 2002. The reactor vessel was removed in one piece from the containment building and loaded into a shielded cask for transport and disposal. However, negotiations broke down with Panama, operators of the Panama Canal, and the vessel could not be shipped through the Canal. Other options were explored, including shipping it around South America to the Barnwell, SC, facility, but no successful solution was developed. The vessel is temporarily stored on the SONGS-1 site.

In 2004, the Rancho Seco nuclear plant in Sacramento, CA, employed mechanical cutting technologies to segment its RV internals. Mechanical cutting eliminated the problems associated with water clarity experienced at other facilities, but the cutting process was very slow. It took more than two years to segment the internals compared with the one year experienced at other plants. The roads and bridges surrounding the Rancho Seco plant were not strong enough to handle the heavily shielded RV in one piece, so abrasive water jet cutting was used again to segment the vessel. They encountered problems with separating the grit from the fine steel particles and collecting the spent grit on the other side of the cut. Nevertheless, the project was completed.

These vessel and internals segmentation programs indicate that major advances have been made in the cutting technologies being used, but significantly more work needs to be done. The lessons learned from each project are valuable input to future designs and applications.

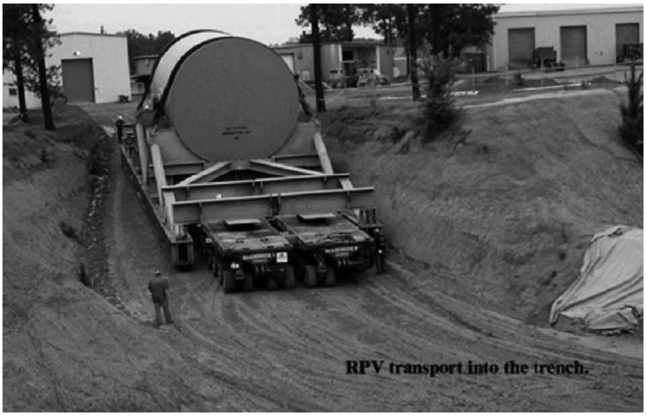
24.2 San Onofre Unit 1 reactor vessel removal.

24.5.4 Heavy transport methods

The transport of one-piece reactor vessels, steam generators, and pressurizers is a major challenge for decommissioning (Fig. 24.3). One-piece removal saves considerable time and cost by avoiding segmentation and its exposure to workers performing the cutting. However, these components are massive. A large 1100 MWe reactor vessel package with shielding weighs more than 907 metric tons; a steam generator from the same plant weighs almost 492 metric tons, and a pressurizer weighs about 246 metric tons. Decommissioning planners must evaluate all possible options depending on the routes available to the disposal facilities.

For road transport, multi-wheel transporters are available with more than 320 separately articulated wheel assemblies to negotiate sharp turns

24.3 Rancho Seco steam generator transport.



24.4 Maine Yankee reactor vessel disposal.

(Fig. 24.4). These transporters can handle loads over 1090 metric tons. They usually have a forward tractor pulling the load and a following tractor to use for emergency braking. Truck transport is generally used for distances less than 1000 miles (1610 kilometers).

For rail transport, depressed-bed railcars are designed to handle loads of up to 180 metric tons. They too are multi-wheel cars to distribute the load over the rails. Heavier loads up to about 1090 metric tons use a Schnabel car, a multi-car assembly with a major frame support for each end of the load and each frame resting on a separate multi-wheel railcar. The load actually becomes part of the support system. This type of railcar is generally used for distances less than 3000 miles (4800 kilometers), but is further limited by the capacity of bridges and the width of tunnels along the route.

For much heavier loads, possibly with multiple components shipped simultaneously, barge transport is used (Fig. 24.5). Sea-going barges can handle many thousands of tons per shipment. Each shipment must meet Coast Guard, naval architect design, and state/federal regulations. Distances are virtually unlimited, as multiple tugs are used to ensure the load is under control at all times. The only limitation is the available depth of water in rivers from the nuclear plant site and the disposal site.

24.5.5 Robotic concrete demolition methods

Concrete demolition is another critical challenge in a decommissioning project. Nuclear-grade concrete is usually the strongest concrete available, testing at more than 55,160 newtons per square meter compressive strength. As concrete ages it continually cures and gets stronger, reaching almost 131,000 newtons per square meter compressive strength. Early reactors used manually held jackhammers but these were found to be ineffective. Later, hydraulically and pneumatically operated demolition hammers were mounted on backhoes and excavators to demolish massive concrete sections. At Shippingport, such hammers were used to demolish the concrete containment structures enclosing the reactor vessel and steam generators.

More recently, companies such as Brokk introduced robotically controlled demolition hammers mounted on a mobile track or wheel chassis. The operator controlled the hammer with a joystick connected by cables to the chassis. That way the operator could remain in a low dose area and away from falling debris without injury. These Brokks can also be fitted with

24.5 Maine Yankee steam generators on barge.

scabbling heads to scarify (remove) the surface layer of contaminated concrete from a low-dose area of the plant. Brokk has recently introduced an articulated robotic arm capable of picking up small parts and components weighing up to 115 kilograms remotely with a joystick. Computerization is advancing this technology.

24.5.6 Advances in controlled blasting

Controlled demolition of concrete structures by blasting (or implosion) has long been used in the wrecking industry for industrial buildings, hotels, stadiums, and other large structures. The technique involves 'softening' the building by removing all interior walls and equipment, and removing the concrete covering of steel reinforcement to place explosives. The explosives are covered with a blanket material to focus the explosive charge inward towards the steel supports and columns. Strategically timed delays are used to fire explosives that cause the building or structure to fall in a specific direction. This manner of demolition drops the building to grade level so additional conventional demolition can be accomplished and rubble can be removed (mucked out). The method greatly accelerates the demolition time and simplifies the cleanup process.

At the Maine Yankee decommissioning project, controlled blasting of the massive containment building was successfully used in 2004 (Fig. 24.6). The containment building was a 4-foot (1.22-m) thick concrete dome structure, heavily reinforced with number 18 reinforcing steel about 2 inches (5.4 cm) in diameter, on 6 inch (15 cm) centers in two layers near the outer and inner

24.6 Maine Yankee containment building demolition.

surfaces of the walls. The inner surface of the containment building also had a 1-inch (2.54-cm) thick steel liner shell. The structure was designed to withstand a major pipe rupture accident within the building, releasing the energy of the high-pressure steam within the reactor system. The building could also withstand the direct impact of a Boeing 747 airplane crash. Conventional demolition using a wrecking ball or demolition hammers would be ineffective and very slow.

To soften the building, arches were cut into the walls of the structure to remove some of the concrete and steel reinforcement. Charges were placed in the remaining 'legs' of the building, and detonated with timed delay fuses. The building dropped in a cloud of dust and rubble, bringing the dome of the building down to grade level where it could be further demolished by conventional methods. The controlled blasting saved many months of conventional demolition, and saved time and money.

Controlled blasting was also used to drop the Maine Yankee turbine building – a more conventional method for its final stage of demolition. The improvements in productivity by this method reduced the overall cost of the project, and led to the timely completion and termination of the site license.

At the Trojan nuclear plant in Oregon, the two huge 46-m hyperbolic cooling towers were also demolished by controlled blasting. The technique is gaining wide acceptance in the industry as more and more structures are demolished safely.

24.5.7 International experience in recycling materials

The US has been fortunate to have several low-level waste disposal sites at Barnwell, SC, Hanford, WA, Beatty, NV, and Clive, UT. These sites were originally open to all waste generators until the Waste Policy Act of 1980 (Public Law 1980), and its Amendments (Public Law 1986). After that time, states were to form regional compacts and be host to as many as 16 new waste disposal facilities. Unfortunately, the additional sites never materialized (except for Texas and its original limited disposal facility, now constructing a federal waste and commercial waste disposal facility). The three existing sites were within their respective regional compacts, and were able to ban the disposal from non-compact state members. Barnwell, SC, remained open to out-of-compact generators, but announced in July 2008 that it was closed to all generators except those states within the compact (South Carolina, New Jersey, and Connecticut).

The federal government constructed several disposal sites for waste arising from the DOE's weapons facility decommissioning projects, but they are closed to commercial waste generators. The Waste Isolation Pilot Project (WIPP) in Carlsbad, NM, was dedicated to dispose of only transuranic wastes from the weapons facilities. A large facility was constructed at the Hanford Reservation in Washington to handle low-level waste only from the Hanford cleanup program. Idaho National Laboratory was selected to store spent nuclear fuel from the Navy's nuclear submarine and surface fleet, and thousands of fuel assemblies are stored there.

Sweden and France have been the leaders in Europe in developing both low-level and spent nuclear fuel disposal sites. They integrated their operational reactor program with the disposal program, thereby ensuring their waste disposal and storage locations. Sweden has developed a smelting facility to melt low-level contaminated steel. The smelting causes the contaminants to float to the surface of the crucible where they can be skimmed off and disposed of, and the steel recycled safely.

Similarly in the UK, the Nuclear Decommissioning Authority (NDA) funded and directed major decommissioning projects at Dounreay in Scotland, and at Sellafield in Cumbria. In Spain, ENRESA, the national waste management company, has developed and used for years the El Cabril facility for medium and low activity waste, where it recently added a very low activity repository for waste produced in the dismantling of the Vandellós I station.

The Canadian government, working with the Canadian utilities (primarily Ontario Power Generation), has been developing its strategy and designs for waste disposal facilities. Currently, low-level radioactive waste is stored at the Radioactive Waste Operations Site at the Bruce NPP in Ontario. Spent nuclear fuel is stored at the reactor sites for about six years, and then transferred to dry storage casks. There are no plans to reprocess spent nuclear fuel in Canada. In March 1998, the Canadian Environmental Assessment Agency (CEAA) Panel reported to the Canadian government that the safety of a geologic disposal concept had been adequately demonstrated. It concluded that the deep geologic disposal of high-level radioactive waste (spent nuclear fuel or solidified nuclear waste) 500 to 1000 meters deep within the stable plutonic rock of the Canadian shield constituted a safe and compliant passive long-term storage option.

Other countries are not so fortunate. The public sentiment against waste disposal facilities has thwarted the effort to site them, and caused utilities to store the wastes on the reactor sites. In an effort to minimize the wastes stored, governments have instituted a mandatory recycling program to decontaminate and reuse these materials in safe applications. In Germany, for example, no waste may be stored on site or at central temporary storage facilities until a dedicated effort has been made to decontaminate them for free-release. Those materials that cannot be decontaminated sufficiently may be placed in these temporary storage facilities. Free-releasable materials such as steel are being recycled to make steel rails for the railroads, and clean or very low-level contaminated concrete is crushed and used for the railroad track beds.

These technological improvements are facilitating decommissioning and making good use of strategic materials.

24.5.8 Improved program management methods

The advent of computer applications to the management of decommissioning programs has been an effective tool to manage project schedules and budgets more closely. Computer programs such as Primavera have enabled managers to provide detailed tracking of individual activities with respect to schedule duration and budget management. Managers developed the project into a Work Breakdown Structure (WBS), a hierarchy of logically sequenced activities, and assigned individual budgets and schedules for each activity. Close tracking of progress against these budgets and schedules allows managers to identify potential problems early and to apply corrective action to prevent overspending or schedule slippage. Computer runs are updated weekly so that problems do not develop unexpectedly.

These computer tools have not only improved productivity through the rapid identification of problems, but provide instant feedback of meeting or missing key milestones. The milestones are often tied to the incentive programs for the project, which elevates the importance of corrective actions.

24.5.9 Greater emphasis on worker and public safety

Project safety is the responsibility of every manager of a major program. Safety programs have been recognized as vital to a successful project, as lost-time accidents can cause serious injuries to workers and delay the project. In the past, project managers would employ perhaps one radiological safety manager and one industrial safety, manager. Now, there are entire departments dedicated to worker safety, performing worker training in all disciplines and 'tool box' safety meetings before each new task is begun.

At DOE programs, every DOC is required to incorporate an integrated safety management program into the overall project program. Safety is measured on the same scale as worker performance and budget management. This principle has been carried into virtually every commercial decommissioning project in the US.

In the same manner, the management of public safety has increased its focus, and citizen advisory groups now participate in meetings on how a program is managed and what methods are being used to accomplish a difficult task.

The reduction in lost-time accidents has been significant. Insurance rates for projects are being reduced, thereby lowering costs. Workers, being more aware of safety, have been conscious of avoiding risks that could lead to accidents, thereby improving productivity. Everyone benefits from these programs and they have proved valuable to the industry.

24.5.10 Radioactive waste management planning

The international experience in radioactive waste management planning is extensive, and varies by country according to the availability of waste disposal facilities. Countries with available waste disposal facilities include, for example, the US, UK, France, Sweden, Spain and Russia. Chapter 14 on spent fuel and radioactive waste management provides a more extensive description of waste management facilities and capabilities in current use. The US is fortunate to have three operating commercial disposal sites, and a fourth one under development. In general, waste management planning begins at the disposal site, and works backwards to the waste generator preparations for packaging and transport.

For those countries with operating disposal sites, generally accepted practices have been established for waste acceptance criteria based on the types of radionuclides, half-lives, concentrations, radiological dose considerations, and hazard/toxicity levels (radioactive mixed wastes). The acceptance criteria establish specifications for disposal containers and waste conditioning (grouting of waste in containers or packaging reinforcement) to prevent long-term subsidence after site closure. Specific types of containers are identified, such as 200-liter drums and intermodal containers, and more recently strong, tight fabric bags are being used. Under special application, acceptable waste disposal packaging includes the intact component itself, such as entire steam generators, pressurizers, and reactor vessels. The generator must demonstrate that these intact components satisfy all the same regulatory requirements as specification containers, or demonstrate that exceptions to the specifications do not endanger public health and safety.

The modes of transport include truck, rail and barge shipments depending on the package characteristics, dose level, weight and physical size. The number of specification containers permissible by truck transport, for example, is limited by international safe transport guidelines established by the IAEA and adopted by virtually all countries. One-of-a-kind intact component shipments are usually transported on a dedicated carrier (truck, rail or barge) and must meet all the same safe transport criteria of a 10-meter drop, drop onto a rigid post, immersion in water, and a 30-minute fire test. Exceptions are permitted if it can be demonstrated there will be no impact on safety.

For those countries without operating disposal facilities, wastes are being stored on site at the nuclear facility, or transported to a central storage facility where additional waste processing and conditioning is performed. For example, in Germany wastes from the nuclear power plants are sent to the Central Decontamination Department at Karlsruhe where various decontamination techniques are applied to separate and free-release materials. The remaining wastes are packed into 200-liter drums and the drums are loaded into half-height intermodal containers and grouted in place. The intermodal containers are transferred to storage until a national repository is approved for ultimate disposal.

In Sweden and in the US, Studsvik has developed a metal melting facility where radioactive metals are melted to separate the radioactivity (which floats to the top with the slag) from the remaining metal. The radioactivity is skimmed off the top and solidified in containers for long-term storage, and the remaining metal can be either free-released or used to fabricate shielding blocks or waste disposal containers.

The disposition of spent nuclear fuel has taken several paths. A detailed description of spent fuel is considered in Chapter 14. Spent fuel is not truly a decommissioning waste, but arises as a significant issue when decommissioning a NPP. The location of the spent fuel in either a wet storage pool or dry casks on site represents a liability that must be accounted for when planning and implementing decommissioning. The site license cannot be terminated if fuel remains on site, and either the license is maintained, or a new spent fuel license is issued.

Fuel reprocessing was the early choice for disposition of the spent nuclear fuel, and the US was a leader in this technology. However, concerns for the proliferation of potential weapons-grade materials shut down this US industry. Nevertheless, there are extensive facilities in France at COGEMA in La Hague, in Great Britain at Sellafield, in Japan at Rokkasho, in India at Kalpakkam, and in Russia at Mayak, successfully reprocessing commercial nuclear fuel. The alternative to reprocessing is temporary on-site storage in dry casks, and then ultimately deep geological disposal. This is the current practice in the US, with each reactor site storing its spent fuel in dry casks. In 1982, Congress passed the Nuclear Waste Policy Act (NWPA 1982) to build a deep geological repository at Yucca Mountain, NV, although work has temporarily been halted as the federal government seeks alternatives to deep geological disposal. There are several lawsuits by states, industry groups and regulator associations charging that the federal government does not have the right to override a Congressional decision to construct the facility. Other countries without reprocessing capability have been shipping their fuel for reprocessing at Sellafield and La Hague, for example, and accepting long-term liability for the residual wastes generated. The old USSR accepted spent fuel from the Eastern European countries and Finland.

24.5.11 Lessons learned from sharing previous experience

The international interest in decommissioning prompted numerous agencies to focus attention on decommissioning in addition to other nuclear operational topics they normally pursued. In the US, the Electric Power Research Institute (EPRI) is an organization funded by member utilities to perform research beneficial to operating utilities. When decommissioning became of greater interest in the 1970s, EPRI dedicated a portion of its staff to address issues valuable to companies approaching ultimate shutdown. EPRI also monitored dismantling progress at several US commercial nuclear power plants and published reports on the lessons learned. EPRI has joined other agencies such as the American Nuclear Society in joint conferences on waste management and decommissioning.

The International Atomic Energy Agency (IAEA) similarly established a waste management and decommissioning section of its staff to provide standards and guidance documents for developing nations to use in creating regulations for each of their countries. The IAEA also sponsored studies on cost estimating and funding, and strategy selection, and in 2005 published two TECDOC reports on these topics: *Financial Aspects of Decommissioning* (IAEA, 2005a). and *Selection of Decommissioning Strategies: Issues and Factors* (IAEA, 2005b). In December 2006, the IAEA sponsored an international conference in Athens, Greece, to provide a venue for the sharing of recent decommissioning experience (IAEA, 2006b).

The Organization for Economic Co-operation and Development/Nuclear Energy Agency (OECD/NEA) has been active in the development of programs for the safe implementation of decommissioning. It celebrated its fiftieth year of operation in 2008, and has been a major facilitator of new and emerging technologies for cost estimating and planning. In September 2004, OECD/NEA sponsored a major international conference in Rome, Italy, for the exchange of recent experience. The OECD/NEA is currently working on programs to explore the differences in decommissioning cost estimates (OECD/NEA, 2010), standardizing decommissioning cost lists of activities, and providing guidance on cost controls.

The World Association of Nuclear Operators (WANO), originally formed to focus on operating reactor performance and programs, has more recently expanded to address decommissioning issues. In May 2005, it sponsored a decommissioning workshop in Malmö, Sweden, to allow the sharing of recent decommissioning experience.

The European Commission (EC) also sponsors research and development of technologies to assist member nations to prepare for decommissioning. It joined forces with the IAEA and OECD/NEA to develop a *Proposed Standardized List of Items for Costing Purposes in the*

Decommissioning of Nuclear Installations (OECD/NEA, 1999), which is now being updated into the *International Structure for Decommissioning Costs* (still in draft form), adopted in the interest of preparing consistent formats and content of cost estimates. The US DOE and consulting companies also contributed to these documents to further expand their applicability.

The ability to share information through these organizations is one of the distinctions that make the decommissioning industry unique. Few other industries in our history have so thoroughly dedicated resources to the advancement of technologies and experiences. It is by this mechanism that the industry can advance the science and improve the safety, lower the cost, shorten the schedule, and reduce the waste generation of these major projects.

24.6 Overview of the decommissioning phase of a nuclear power plant (NPP) lifecycle

An owner/licensee's decision to shut down a NPP is often driven by more than one factor, and the evaluation needs to consider the impact of all the factors in a cost–benefit analysis. The major factors include the end of the license life, the economic viability of continued operation, technological operating problems, and socio-political issues. These factors will be discussed in the following paragraphs.

- *End of license life*. Clearly the simplest driver is the end of the license life. Most NPPs are licensed for a period of about 40 years, which was established based on the typical life of a fossil-fueled power plant, and more importantly on the expected structural integrity of the reactor vessel after 40 years of neutron bombardment. Vessel steels begin to lose ductility when exposed to the neutron flux (dose) from radial core leakage beyond the core boundary. Periodically, vessel material samples are removed from the vessel and subjected to a Charpy Impact Test which provides a measurement of the Nil Ductility Transition Temperature (NDTT). The NDTT is the temperature to which the vessel must be heated before pressurization to prevent brittle fracture of the metal. When the NDTT equals or exceeds (within a safety margin) the normal operating temperature of the reactor, the vessel is deemed unsafe for continued operation and the plant must be shut down.

The technology of reactor vessel manufacturers has advanced to enable the vessel to be annealed by reheating the vessel in place to regain some ductility, but this has not been tried commercially. With the recent decommissioning experience in the US and internationally, it may also be possible to remove and replace the reactor vessel and continue operating for potentially another 40 years. But again this has not been attempted commercially.

To date the common practice has been to shut down the reactor and decommission the NPP. The evaluation of the other drivers usually compounds the decision to decommission the plant. It has been clearly established that pressure vessels in currently operating nuclear power plants can safely operate for 60 years, or more, by establishing a complete and satisfactory age monitoring program. The lives of current operating plants may be extended to 60 years and current nuclear designs offer an operating lifetime of 60 years.

- *Economic viability of continued operation.* The economic viability of continued operation depends on the cost of competing power sources in the region or connected to the power grid, and the cost of operating the NPP which includes fuel, labor, equipment replacement, taxes, insurance, permits, and radioactive waste disposal. Presumably, competing power sources would represent newer NPPs, fossil-fueled power plants, or alternative technologies of wind or solar, which could be less costly to operate (although to date that has not been the case). The cost for nuclear fuel is currently a stable commodity, but future costs will depend on the marketplace and need to be evaluated at the time the plant nears the end of its life. Further, the cost for spent fuel storage, reprocessing or disposal needs to be factored into the cost–benefit analysis. Labor costs are generally driven by inflation, by trade union agreements, and by the numbers of personnel needed to operate the NPP. Equipment replacement costs become a major determinant for continued operation, as replacement of the steam generators of a pressurized water reactor represents a cost of about $500 million for a 1200 MWe NPP. The safety systems actually are maintained in virtually perfect operating condition as they are expected to have to perform their full safety function up until the very last day of operation in the event of an accident. Taxes, insurance and permits are essentially overhead costs that are fairly stable over the operating life of the plant. Radioactive waste disposal costs increased rapidly over the first 25 years of the current generation of operating nuclear plants, but have generally stabilized over the last 10 years due to market pressures, waste reduction technologies and alternative waste processing methods.

Each of these factors needs to be considered in cost–benefit analyses before making decisions of shutting down a NPP for decommissioning. Risk assessments have become a major part of such analyses, as they provide guidance in 'what-if' scenarios addressing future continued operation. Several NPPs in the US, UK, France, Sweden, Spain and Germany were shut down for economic considerations. In Germany the Neckar-Westheim plant and in Spain the Vandellós 1 were closed because the cost of refurbishing such plants to comply with regulatory requirements was too expensive. In Italy nuclear power plants were closed after a national referendum.

- *Technological operating problems.* Technological operating problems were prevalent in the early designs of NPPs in the 1950s and 1960s, as new technologies were tested and shown to be non-commercial or unsafe. The current generations of NPPs have been through the redesign process and have implemented operating cost reductions so they can compete commercially with fossil, wind, and solar energies. Operating problems such as leaking steam generators, fuel leakage problems, reactor vessel head corrosion, spent fuel storage capacity, and radioactive waste disposal/storage have been dealt with as an operating expense. The economic viability of NPPs in the current power market has shown that the equipment replacement costs are small relative to the electricity generating capacity and potential profits from continued operation. Only when a plant nears the end of its license life (or extended life in some cases) do technological operation problems become an issue for consideration of shutdown.
- *Socio-political issues.* The socio-political issues of a NPP shut-down can represent a significant impact on the local community. In some cases the local anti-nuclear sentiment was strong enough to drive the decision to shut down a viable NPP. A prime example of this was the Shoreham NPP in New York which only operated for three Effective Full Power days. The local concern of the potential for an accident with very limited emergency escape routes on overcrowded roads was sufficient for the owner/licensee to shut down and decommission the unit prematurely. On the other hand, shutdown of a large NPP will put hundreds of workers out of jobs and affect the viability of the local community, causing a downward spiraling of the local economy. This impact is discussed more completely in the IAEA report, *Financial Aspects of Decommissioning* (IAEA, 2005a).

24.6.1 Pre-decommissioning activities

Once the decision to shut down a NPP has been made, there are a number of activities that must be performed to prepare for decommissioning. These activities include selecting the decommissioning strategy, determining the facility end point, establishing criteria for release of the facility, preparing regulator documents, identifying long-lead issues, staff training, and preparing a detailed cost and schedule estimate with a risk assessment. These activities will be discussed in the following paragraphs.

There are three generally accepted decommissioning strategies: immediate dismantling, safe enclosure, and entombment. The IAEA defines these as follows.

- *Immediate dismantling*. Immediate dismantling commences shortly after shutdown (normally within 5 years).This period is known as the transition period necessary to prepare for decommissioning. Decommissioning is expected to commence after the transition period and continues in phases or as a single project until release of the facility or site from regulatory control.
- *Safe enclosure.* Decommissioning may be deferred for a period of up to 100 years. Safe enclosure is a strategic option in which a facility or site is placed in a safe condition to allow for delayed decommissioning. During the safe enclosure period, a surveillance and maintenance program is implemented to ensure that the required level of safety is maintained. During the shutdown and transition phases, facility-specific actions are necessary to reduce and isolate the source term (removal of spent fuel and conditioning of waste) in order to prepare the facility/site for the safe storage period.
- *Entombment.* Entombment is a strategy in which the remaining radioactive material is encapsulated on site. A waste repository is effectively established and the requirements and controls for the establishment, operation and closure of waste repositories are applicable. Surveillance and maintenance, normally at a much lower level than in the case of safe enclosure, are required after entombment. Entombment as a decommissioning strategy is not commonly adopted.

The decision on a decommissioning strategy must also be based on an evaluation of several factors. The IAEA report *Selection of Decommissioning Strategies: Issues and Factors* (IAEA, 2005b). addressed the major issues to be considered in selecting a strategy for decommissioning. These issues include the following.

- National policies and regulatory framework
  – Existence of policy documents that address the regulation of the nuclear industry on a national level
  – Existence and extent of a legal framework covering a regulatory function and infrastructure as well as requirements and standards pertaining to decommissioning
  – Authorizations/licenses including processes to ensure regulation of the full lifecycle of the facility including regulations for the planning and execution of decommissioning
- Financial resources/cost of option
  – Availability of financial resources
  – Cost associated with a specific decommissioning option under consideration
  – Rate at which financial resources will be released and made available for decommissioning

- Spent fuel and waste management system
  - Existence of a national waste management policy and strategy
  - Availability of facility-specific waste management plans
  - Availability of waste management facilities, namely storage, processing and disposal facilities
  - Availability of spent fuel management strategy, plan and facilities
- Health safety and environmental (HSE) impact
  - Safety/health risk of the decommissioning option
  - Environmental impact of the decommissioning option
  - Transport impact of the decommissioning option
- Knowledge management and human resources
  - Physical status of the installation – integrity of buildings
  - Radiological characteristics and the impact of a specific decommissioning option on the radiological characteristics of a specific installation
  - Availability of suitably qualified and experienced personnel
  - Availability of suitable decommissioning technology and techniques
  - Lessons learned from previous decommissioning projects
  - Operational history and lack of information
  - Existence of other operating nuclear facilities on site
  - Reasons for plant shutdown
- Social impacts and stakeholder involvement
  - Social impact of decommissioning option
  - Public concerns and perceptions
  - Reuse options, expectation and demands.

The key to selecting a decommissioning strategy is to understand the radiological, hazardous, and physical inventory of the materials present on site. This is accomplished by conducting a thorough site characterization program of the facility and site. The program includes a historical site assessment (HSA) of the activities conducted during operation to predict the extent and levels of contamination and activation in the facility and site. Sampling and surveys are conducted using approved instrumentation in accordance with sampling protocols approved by the regulator, and data are maintained under a qualified quality assurance program. Data are analyzed by qualified radiological and hazardous waste professionals and a final report is issued for all potential users involved in the planning and implementation of the decommissioning project. these characterization data form the basis for all cost estimating, planning, decontamination, dismantling, waste packaging, transport, and disposal. The characterization data may be supplemented as additional information is uncovered as the field implementation is conducted. The following aspects need to be considered:

- *Determination of facility/site end points.* Decommissioning strategy selection must consider the facility and site end points desired. Whether the facility and site are to be free-released for reuse (greenfield) or retained under administrative control (brownfield) are key elements in selecting a strategy. There are no universally accepted definitions of greenfield or brownfield, and the specific conditions to be achieved for either of these end points must be clearly identified at the outset.
- *Establishment of the criteria for the release of the facility and site.* Whether the strategy involves a greenfield or brownfield end point, the criteria for release of the site under each strategy should be identified at the outset, and all parties and stakeholders made aware of this decision and its implications.
- *Regulatory document preparation.* All regulatory documentation notifying the regulator of the intention to decommission needs to be filed during the pre-decommissioning phase. Each country's regulator may have specific requirements to be observed in preparing such documentation.
- *Long-lead issues of planning/contracting.* Decommissioning planners need to identify the long-lead issues, equipment and contracting requirements to have these resources available when needed in a timely manner.
- *Staff training.* In many cases, the owner/licensee will play an active role in the decommissioning of the facility. Few such operations personnel have the necessary detailed training for such work, and it is important to arrange for formal training courses either at the NPP facility or off-site at a training facility.
- *Preparation of detailed cost estimate, a schedule and risk management program.* During the pre-decommissioning phase, a baseline cost estimate, schedule, and risk assessments should be established. These documents should be living documents, adjusted and updated periodically as the program evolves and new information or changing conditions occur.

24.6.2 Transition phase

The transition phase of decommissioning is the period during which the management organization and facility change over from operations to decommissioning. The major issues involve identifying key personnel and retaining them for decommissioning, reassigning or terminating redundant personnel, removal of spent fuel to on-site storage, preparing the facility and site for decommissioning, and completing the site characterization program. These are discussed in the following paragraphs.

- *Key personnel retention issues.* The owner/licensee will likely participate in the detailed decommissioning activities to some extent, and accordingly will need to identify key personnel to be retained until major milestones are accomplished. Early identification is important to ensure these personnel will not seek employment elsewhere, and incentive plans are generally offered to secure their service for the needed duration.

- *Redundant personnel issue.* Redundant personnel should be notified early so they can make preparations for future employment. Owner/licensee out-placement assistance can be a valuable tool for a smooth transition for these employees.
- *Spent fuel removal to on-site storage.* Transfer of the spent fuel to on-site storage (wet or dry) facilities may be accomplished during this period. Depending on the type of system used, this may require more than one year to accomplish, and may be on the critical path of major activities.
- *Draining and securing systems.* All non-essential (to decommissioning) systems can be drained and secured at this time. The operating plant staff is generally used for this work.
- *Preparing the site and facility for decommissioning.* To facilitate the increased flow of vehicular traffic during decommissioning, changes to the security fencing and number of gates may be modified and increased. Additional security equipment, vehicular radiological survey equipment, weighing scales, and closed circuit TV (CCTV) systems may be installed.
- *Site radiological and hazardous material characterization.* Continuation of the facility and site radiological and hazardous material characterization program should be conducted at this time. As noted earlier, characterization is a continuing activity as new conditions arise and additional information is uncovered.

24.6.3 Contracting phase

In many cases the owner/licensee may decide to use the services of an experienced decommissioning operations contractor (DOC) to manage the decommissioning activities. The special management experience of a DOC can be a major asset to a successful and cost-beneficial program. Selecting a contractor consists of soliciting bids, evaluating bids, contract negotiations, award of contract and mobilization. The following paragraphs describe these activities.

- *Preparation of decommissioning operations contractor (DOC).* The owner/licensee must prepare bid specifications identifying the scope of work intended for the DOC, the schedule for performance, terms and conditions, and the type of contract to be awarded. Typical contract types include:

- *Time and materials.* The DOC is paid its standard cost plus fee for each individual for the hours worked, plus the cost of expenses for office space, utilities, travel, and other miscellaneous items.
- *Fixed-price (lump sum).* The DOC is paid a fixed amount (sometimes called a lump sum) for the complete project (paid upon completion of milestones). No change orders are generally permitted unless the work scope changes.
- *Cost-plus-fixed fee.* The DOC is paid all of its costs for labor and materials, plus a fixed fee (profit) based on successful delivery of the work scope.
- *Cost-plus-incentive fee.* The DOC is paid all of its costs for labor and materials, plus an incentive fee for the work-scope performed by a specified delivery date, and a penalty for late delivery. Incentive fees are generally higher than a fixed fee, and penalties are proportionately severe.

In each of these cases it is important for the owner/licensee to carefully identify the work scope and get written agreement with the bidder as to what is desired. Failure to follow this work-scope identification usually ends in costly and time-consuming litigation.

- *Solicitation of bids and evaluation of bidders.* The owner/licensee should secure three or more qualified bidders for the solicitation. This process may require six months to a year to accomplish, as the bidders need sufficient time to analyze the project and prepare their bids. The bid evaluation process itself can take an additional six months for a fair assessment. In some cases, the owner/licensee may secure the services of an independent consultant to evaluate the bids on a consistent basis.
- *Selection of bidder and contract negotiations.* Bidder selection is followed by negotiation of the contract terms and conditions, contract pricing, delivery schedule, and proposed organization personnel assigned to the project. These negotiations may require six months to accomplish.
- *Early mobilization of contractor.* The successful DOC will require several months to mobilize his staff (and in some cases, heavy equipment needed for the project) on site. Getting adequate telephone service for the DOC staff can sometimes be a challenge in itself. Purchasing computers, printers, fax machines, and getting the staff familiar with their use is necessary 'non-productive' time.

24.6.4 Decommissioning planning by contractor/owner-licensee or DOC

On-site decommissioning planning can require as much as one year to prepare all the program documentation and to secure approvals for the work. This planning effort includes an organization plan, personnel assignments, project management planning, detailed procedures, work plans and permits, implementing facility and site modifications, and equipment purchase or leasing. The DOC may already have prepared some of these documents in its proposal. These activities will be described in the following paragraphs.

For the purposes of the following descriptions, the immediate dismantling strategy will be assumed as it is the most complete of the possible strategies. safe enclosure or entombment are deferral strategies, but ultimately will require some level of dismantling similar to immediate dismantling.

- *Development of organization plan.* The project organization plan identifies how the project organization will be managed, and is generally linked to the work breakdown structure (described later) of the activities to be performed. The organization plan provides the functional relationship of each of the management positions and their roles and responsibilities.
- *Personnel assignments.* Once the organization plan is developed, personnel assignments can be made and hiring (as needed) initiated. As noted earlier, key personnel will be retained for specific work as tied to the milestones of the project.
- *Project management planning.* Management planning includes all those activities such as a project policy manual, quality assurance manual and procedures, radiological safety and health physics policies and procedures, industrial safety policies and procedures, cost and schedule control system, risk management system, purchasing policies and systems, engineering procedures, and documentation and drawing controls.
- *Preparation of detailed procedure.* Detailed procedures describe what work is to be performed and how to accomplish the work. Specific guidance is included as to where to make cuts in piping, how to remove major components, how to de-energize electrical equipment, or how to demolish concrete structures. Prerequisites and safety provisions are identified, and a checklist of completed steps is provided. All necessary approvals are identified before any fieldwork can be performed.
- *Preparation of work plans and permits.* Work plans and permits are the final documentation needed before fieldwork may begin. They include specific radiological controls, personnel protection, industrial safety controls, fire permits (for thermal cutting, for example), confined entry permits, and area ventilation controls.
- *Facility and site, and security modifications for decommissioning.* At this time, all facility and site modifications and security modifications may be completed. These include additional change rooms, laundry facilities, filtered air-mask cleaning stations, and waste management area arrangements for transport and disposal.

- *Equipment purchase/leasing.* Major equipment such as cranes, front-end loaders, fork lifts, and specialized equipment such as vessel and internals cutting equipment, are long-lead items. They will require purchase specifications, bidding, award and contracting in a timely manner to have the equipment available when needed. Purchase or leasing decisions need to be made based on a cost–benefit analysis for each major purchase.

24.6.5 Decommissioning and dismantling activities

The decommissioning and dismantling phase represents the physical work associated with characterizing the facility, then removing and dispositioning the equipment and structural materials for disposal.

- *Verification of characterization of facility and site.* As noted earlier, characterization is an on-going activity as conditions change during decommissioning and new information is uncovered. During the decommissioning activities, the characterization information obtained earlier should be verified and updated.
- *Removal of redundant systems and structures.* All non-essential (to decommissioning) redundant systems and structures may be dismantled and removed. These include the turbine-generator, condenser, feedwater and condensate systems, high-pressure and low-pressure safety injection systems, and portions of the component cooling water systems not related to building air conditioning. Material that can be removed opens up additional areas for laydown of components and piping in preparation for packaging and transport for disposal.
- *Removal of the large components (steam generators, pressurizers, reactor coolant pumps).* The large components such as the steam generators, pressurizers and reactor coolant pumps can be removed at this time. These are major components, and each will require engineering analyses, rigging analyses, packaging approvals, transport approvals and disposal arrangements. The large size of these components requires multi-wheel transporters, and favors barge or rail transport when available, or they may be further segmented to be handled by truck transport.
- *Removal of the reactor vessel internals.* Removal of the reactor vessel internals must be accomplished by remote or semi-remote methods because of the very high dose rates emanating from the internals. Specialty contractors are usually retained for this work, which have the tooling and experience to perform the cutting operations. Cutting methods such as plasma arc torches, high-pressure abrasive water jet cutting, and mechanical cutting have been used successfully for this

work. In some cases such as in the Trojan NPP in the State of Oregon, the owner/licensee was permitted to leave the internals in place and remove both the vessel and internals intact for disposal. But these are one-of-a-kind evolutions which require special regulatory approval.

- *Removal of the reactor vessel.* The reactor vessel head is removed, segmented if necessary and transported for disposal as low-level radioactive waste. The reactor coolant piping is disconnected from the vessel by cutting with milling cutters or diamond wire saws, and disposed of as radioactive waste. The other reactor vessel appurtenances (in-core instrumentation, level indicators and water sampling piping) are removed and disposed of as radioactive waste. The reactor vessel may be grouted in place if the existing carriage can be modified to handle the extra loads, or lifted from the reactor cavity and loaded into its shielded transport container outside the containment building. Grouting may then be performed to secure the vessel in the container and to provide additional shielding as required. The vessel is usually transported by multi-wheel transporters on site to a barge facility or rail siding for transport to the disposal facility.
- *Demolition of radioactive structures.* All remaining radioactive structures or potentially radioactive structures can be removed at this time. These include the biological shield, supporting structural steel and concrete, and all contaminated concrete may be demolished by conventional methods. In some cases, controlled blasting has been used successfully for massive concrete sections. Otherwise, hydraulic rams mounted on a backhoe (a ram-hoe), or an excavator have been used.
- *Survey and sampling for determination of license termination requirements.* Upon the removal or the last of the radioactivity, the final survey and sampling program may be initiated to certify that the license may be safely terminated. All necessary documentation is submitted to the regulatory agencies, and the regulatory agency's independent verification contractor confirms the license termination requirements have been met. The license may then be terminated.
- *Demolition of structures.* Demolition of the remaining structures such as the containment building, auxiliary building, turbine building, diesel-generator building and intake and discharge structures may be accomplished by conventional methods. In some cases, controlled blasting has been used as was done at the Maine Yankee NPP in the US, to lower the containment dome so that conventional hydraulic ram-hoe methods could be used.
- *Restoration of site.* The remaining site may be restored to either a greenfield or a brownfield condition. As noted earlier, there are no universally accepted definitions of these terms, and a pre-agreed condition must be in place before work begins.

24.7 Management of decommissioning waste and the recycling of materials

The key to successful decommissioning is the planned and organized management of waste disposition and recycling of materials. All the decommissioning planning, technologies, and implementation activities are futile if the disposition of the waste is not properly addressed. As noted earlier, detailed decommissioning planning needs to start at the waste disposal or storage facility, and worked backward to the NPP for planning how to meet the requirements and waste acceptance criteria of the disposal or storage facility. This section will discuss the classes of waste, sources and waste streams, and packaging, transport and disposal or storage.

24.7.1 Decommissioning wastes

The disposition of wastes requires conformance to the waste acceptance criteria (WAC) of the disposal or storage facility. The WAC involves classifying the waste, identifying the waste streams, quantifying the concentrations and volumes or weights, and the means for packaging, transport and disposal or storage.

Classes of waste

Waste classifications are generally well established internationally. There are six accepted classifications: Exempt Waste, Very Short Lived Waste, Very Low Level Waste, Low Level Waste, Intermediate Level Waste, and High Level Waste. These are defined by the IAEA in its General Safety Guide GS-G-1 (IAEA, 2009).

By comparison, the US Nuclear Regulatory Commission identifies only three levels of low-level waste, Classes A, B and C. These waste classifications are identified in Title 10 of the Code of Federal Regulations, Part 61.55 (US NRC, 2011), and are classified by specific radionuclide and concentration:

- In simple terms, Class A waste is waste that is usually segregated from other waste classes at the disposal site. If Class A waste also meets certain stability requirements, it is not necessary to segregate the waste for disposal.
- Class B waste is waste that must meet more rigorous requirements on waste form to ensure stability after disposal.
- Class C waste is waste that not only must meet more rigorous requirements on waste form to ensure stability but also requires additional measures at the disposal facility to protect against inadvertent intrusion.

- Waste that is not generally acceptable for near-surface disposal is waste for which form and disposal methods must be different, and in general more stringent, than those specified for Class C waste. This waste is called Greater-Than-Class-C (GTCC) wastes. This waste is likely to be disposed of as High-Level Waste.

The US NRC identifies high-level waste as the wastes from reprocessing of spent nuclear fuel, which will ultimately be disposed of in a deep geological repository. Spent nuclear fuel is currently stored on site at the NPPs in dry casks awaiting a deep geological repository. Transuranic wastes are from the US nuclear weapons complex and are currently disposed of at the Waste Isolation Pilot Plant (WIPP) near Carlsbad, New Mexico.

In Title 10 of the Code of Federal Regulations, Part 20 (US NRC, 2007), the US NRC may grant free release of radioactive waste on a case-by-case basis, provided the activity does not exceed 1 mrem/year (0.01 mSv/year). Such waste may be disposed of in a landfill.

Sources of waste – contamination and activation

There are two sources of radioactive products at a NPP: fission and activation. Fission and activation are the two nuclear reactions which may produce radioactive isotopes; these radioisotopes may contaminate other materials or become part of them and so produce radioactive waste. Radioactive fission products may be released from leaking fuel elements, including short-lived radionuclides, as gases of xenon-133 and 135, and iodine-129 and 131, strontium-90, cesium-137 and transuranics. From a decommissioning standpoint, xenon and iodine will be essentially gone by radioactive decay by the time decommissioning begins. However, strontium, cesium and transuranics will remain as contamination on the internal surfaces of the reactor vessel and internals, steam generators, pressurizers and reactor coolant piping. Another source of contamination is from minute particles caused by erosion of the metallic materials used in the reactor coolant system which are irradiated by the core as they pass through the system. These particles (sometimes called CRUD from their discovery at the Chalk River Laboratory in Canada as an Undefined Deposit) are also deposited on the interior surfaces of systems in contact with the reactor coolant system. They may also be observed on the exterior surfaces from leakage or spillage of the coolant. Contamination levels may be reduced or eliminated using various decontamination solvents or chemical processes.

Activation is associated with the reactor vessel and internals from direct neutron bombardment from the core. The radioactivity is distributed throughout the material as opposed to collecting on the surface, and in general reduces in source strength farther from the core. The biological shield surrounding the vessel is usually activated to one or more meters from the interior surface, to levels exceeding free release. There is no known method for decontaminating activated materials.

Waste streams

From a decommissioning standpoint, waste streams originate from both contamination and activation. In general, the waste streams are (1) contamination: piping and components, electrical cable, conduit and switchgear, concrete and steel structures, soils, groundwater, dry active wastes, protective clothing, and filters; and (2) activation**:** reactor vessel and internals, and the biological shield.

Packaging, transportation and disposal/storage

Waste management involves the selection and use of packaging of the radioactive materials for safe transport for storage or disposal. There are numerous documents from the IAEA on the various packages for waste disposal from decommissioning. Suffice it to say that most waste materials are packaged in 200-liter drums or in full-size or half-size intermodal containers for transport and storage or disposal. Exceptions are the reactor vessel and internals which require a specially designed container for the vessel, and a licensed shielded cask for the internals for transport with a disposable liner. Container selection, and transport methods and safety measures require an engineered analysis of the radiological contents based on the characterization information, and the routes and modes of transport to be employed.

For those countries with an approved operating disposal facility, disposal is the obvious option. For those countries without a disposal facility, temporary storage is the likely option. As discussed later, waste processing for free-release or restricted reuse is an alternative to direct disposal or storage.

24.7.2 Recycling wastes

The limitations on available disposal space make recycling and reuse a valuable and cost-effective means for disposal. To take advantage of this approach, accepted criteria for free-release must be agreed upon, and proven methods used to decontaminate materials so they may be safely recycled. This section describes these considerations.

Criteria for free-release

When considering free-release of materials, the radiological status of the material can be determined from the characterization program data.

However, the criteria for free-release are usually a regulatory function often subject to strong stakeholder input and acceptance. Many a decommissioning program has been halted or delayed because of the lack of proper communication of free-release criteria to be used. That is why it is important to establish the criteria at the outset of the program to avoid last-minute stakeholder intervention.

Segregation and free-release

One of the lessons learned from recent decommissioning programs is the use of off-site waste processors. These contractors accept waste at their facilities and further segregate the materials for decontamination or direct disposal, or free-release. The cost–benefit of these waste processors has been shown to be very effective in reducing the amount of waste to be buried.

Decontamination and free-release

Early decommissioning programs endeavored to perform decontamination on site to achieve free-release of materials, but they were often ill-equipped to achieve these objectives and resorted to direct disposal. However, more recent experience has shown that certain decontamination processes are effective in free-releasing materials. In particular, electropolishing (reverse electroplating) has been successfully performed at the Gundremmingen NPP in Germany. More than 300 metric tons of material was free-released by this process. It should be noted that in Germany, the law requires the licensee to attempt decontamination for free-release before it is permitted to dispose of radioactive waste at a burial facility.

Smelting, segregation and free-release

Another waste processing technique that has achieved some success is smelting of radioactive metals for either free-release or restricted reuse. In Sweden and now in the US, Studsvik has developed a smelting facility to melt surface-contaminated metals and segregate the contamination in the slag on the top surface of the melt. The slag is skimmed off the top and the remaining material may be either free-released or reused as shielding blocks or waste packaging.

Criteria for the applications for reuse

Reuse of salvageable equipment is another way to reduce waste disposal. The materials must be certified as non-radioactive by the most stringent of regulatory controls to prevent inadvertent release of these materials to the public for reuse. Typically, a 100% survey of all surfaces is required to certify a piece of equipment or material for reuse. Such items as diesel-generators, pumps, motors, switchgear, cranes, and fans are candidates for reuse. However, in general their resale value may be less than 10% of the original purchase price. A cost–benefit analysis should be performed to determine whether the cost to remove a component carefully for reuse will generate sufficient return on investment to cover the cost of removal.

24.8 International experience

The large number of power reactors worldwide has produced a wealth of valuable experience in decommissioning. It is not possible to cover all of this experience in this book, but some of the major contributions will be described herein.

24.8.1 North American experience

As noted earlier, in the US the demonstration reactors of the late 1950s and early 1960s provided valuable lessons in operating characteristics for this developing technology of nuclear power. These plants, including Hallam in Nebraska, Piqua in Ohio, BONUS in Puerto Rico, Elk River in Minnesota, and Saxton, and Shippingport in Pennsylvania, were the early training ground for the commercial nuclear program in the US. But as each of these reactors completed its intended research or demonstration technology, it was shut down and decommissioned. Decommissioning also provided valuable lessons learned in decontamination, vessel and internals cutting technology, concrete cutting and demolition, concrete blasting, and building entombment for long-term safe storage. The technologies used today in recent decommissioning programs are extensions of this early technology augmented by the advent of computer controls, larger-scale demonstrations and heavier equipment lifting and transport methods.

In Canada, the Douglas Point NPP PHWR was shut down in 1984 and placed in safe enclosure. Similarly, the Gentilly Unit 1 NPP BWR was shut down in 1982 and also placed in safe enclosure. Both of these plants were early versions of the CANDU reactor design.

In North America, larger NPPs were decommissioned for various reasons and the experience learned from each project served to provide a technological basis for planning and implementing the subsequent project. A summary of this experience is shown in Table 24.1.

24.8.2 European experience

In a similar manner many of the advanced European countries followed the same pattern of development, building and testing new reactor designs

Table 24.1 Decommissioning experience in North America

Country	Location	Reactor type	Operative life	Decommissioning phase	Dismantling cost
Canada (Québec)	Gentilly-1	CANDU-BWR, 250 MWe	180 days (between 1966 and 1973)	'Static state' since 1986[4][5][6]	Stage 2: US $25 million
Canada (Ontario)	Douglas Point	CANDU-PHWR, 220 MWe	16 years (between 1968 and 1984)	Safe enclosure	?
Canada (Ontario)	Pickering NGS-A2, A3	CANDU-PWR, 8 × 542 MWe	30 years (from 1974 to 2004)	Currently in 'cold standby'– decommissioning in 2012?	(Calculated: $270–430/kWe)
USA	15 reactors	BWR, PWR, FBR, HTR, 3–257 MWe	1–18 years (1963–1987)	Immediate dismantling, entombment, safe enclosure	
USA	Fort St Vrain	HTGR (helium-graphite), 380 MWe	12 years (1977–1989)	Immediate dismantling	$ 195 million
USA	Millstone 1	BWR, 641 MWe	28 years (1970–1998)	Partial immediate dismantling	
USA	Rancho Seco	PWR, 913 MWe	12 years (closed after a referendum in 1989)	Delayed dismantling after 10 years, partial dismantling completed in 2009	$529.6 million
USA	Three Mile Island 2	Multiunit: 913 MWe PWR	*incident*: core meltdown (in 1979)	Post-defueling, Phase 2 (1979)	$805 million (estimated)
USA	Shippingport	(First PWR) 60 MWe	25 years (shut down in 1980)	Dismantling completed in 5 years (first commercial reactor)	$98.4 million

Table 24.1 Continued

Country	Location	Reactor type	Operative life	Decommissioning phase	Dismantling cost
USA	Trojan	PWR, 1180 MWe	17 years (closed in 1993) (excessive cost of steam generator replacement)	Partially dismantled by 2004 (cooling tower dismantled 2006)	\$429 million
USA	Yankee Rowe	PWR, 185 MWe	31 years (1960–1991)	Dismantling completed (greenfield open to visitors)	\$636.4 million
USA	Maine Yankee	PWR, 860 MWe	24 years (closed in 1996)	Dismantling completed – demolished in 2004 (greenfield open to visitors)	\$635 million
USA	Connecticut Yankee	PWR, 590 MWe	28 years (closed in 1996)	Dismantling completed in 2007 (greenfield open to visitors)	\$820 million
USA	Big Rock Point	BWR, 67 MWe	35 years, (closed in 1997)	Dismantling completed in 2007	\$430.8 million
USA	San Onofre-1	PWR, 410 MWe	24 years (closed in 1999)	Dismantling partially completed in 2008	\$622 million
USA	Exelon-Zion 1 and 2	PWR – Westinghouse, 2 × 1040 MWe	25 years (1973–1998) (incident in proceedings, abandoned because of the excessive cost of steam generator replacement)	Safe enclosure – energy solutions contracted to dismantle (opening of the site to visitors for 2018)	\$900–1100 million (2007 dollars)

and molding a framework of nuclear power generation. Some of these designs were successful and some were not, as was experienced in the US. Table 24.2 summarizes the status of operating reactors and those shut down for decommissioning. The lessons learned from this experience have advanced the technological knowledge internationally.

24.8.3 Asian experience

As the growth of nuclear power expanded internationally, Asian countries joined in the development process. Some of the same technologies were used, with variations based on improvements from the earlier experience in other countries. Eventually, these early reactor designs were determined to be redundant and the NPPs were shut down for decommissioning. Table 24.3 summarizes the Asian reactors undergoing decommissioning.

24.9 Sources of further information and advice

There is a wealth of information on decommissioning published by international agencies, government organizations and professional societies. Thanks to the Internet, much of this information is readily available and accessible. Historical records and documents provides a means for determining lessons learned, and recent current experience provides a basis for state-of-the-art engineering and planning. Prudent planners will avail themselves of this information before embarking on a major decommissioning project. This section describes some of the sources of this information.

24.9.1 International Atomic Energy Agency (IAEA)

Sources of additional information are available from a number of agencies. The IAEA, a member of the United Nations, is widely recognized by Member States, nuclear regulatory organizations and the nuclear industry. It publishes documents on virtually all aspects of nuclear energy, including decommissioning of NPPs. The IAEA Safety Standard Series is published in three categories: Safety Fundamentals, Safety Requirements, and Safety Guides.

- *Safety Fundamentals* provide the objectives, concepts and principles of protection and safety and provide a basis for safety requirements.
- *Safety Requirements* establish the requirements to be met to provide protection of humans and the environment.
- *Safety Guides* provide recommendations and guides on how to comply with the safety requirements.

Table 24.2 Nuclear decommissioning in Europe

Country	Location	Reactor type	Operative life	Decommissioning phase	Dismantling cost
Armenia	Metsamor 1	VVER-440, 376 MWe	13 years (1976–1989)	Safe enclosure	
Austria (nuclear-free country)	Zwentendorf NPP	PWR, 723 MWe	Never activated, after referendum in 1978	Used for spare parts for German reactors	
Belgium	Mol	PWR (BR-3)	25 years (1962–1987)	Dismantling completed – pilot project (underwater cutting and remote-operated tools)	
France	Brennilis	HWGCR, 70 MWe	12 years (1967–1979)	Phase 3 – Dismantling	€480 million
France	Bugey-1	UNGG, 540 MWe Gas-cooled, graphite moderator	22 years (1972–1994)	Postponed	
France	Chinon A-1,2,3	Gas-graphite, 1-70 MW–2-210 MW	20 years (1973–1990)	Postponed	
France	St Laurent A 1,2	Gas-graphite, 480 MW–515 MW	20 years (1970–1992)	Postponed	
France	Superphénix at Creys-Malville	Fast breeder nuclear reactor (sodium-cooled), 1200 MWe	12 years (1986–1998)	Postponed	Estimated for the future: \$4000/kWe
France	EL-49 Monts d'Arrée	Water-cooled gas reactor, 70 MWe	15 years (1970–1985)	Postponed	
France	G-1,2,3 Marcoule	Gas-cooled reactors, 3–38 MWe	14–24 years (1970–1984)	Postponed	
United Kingdom	Berkeley	Magnox, 2 × 138 MWe	27 years (1962–1989)	Safe enclosure: 30 years (internal demolition)	\$2600/kWe

United Kingdom	20 reactors	Magnox, 50–235 MWe	50 years (1956–2006)	Safe enclosure: 30 years	
United Kingdom	Sellafield–Windscale (note: Windscale: britain's biggest nuclear disaster)	Windscale advanced gas reactor (WAGR), 32 MWe	18 years (1963–1981), fire of graphite in moderation bars inside the reactor, partial meltdown of fuel[40]	Remote cutting of reactor in 2009 – pilot project (cutting with remote-controlled robots, UV lasers)	More than $2600/kWe (WNI estimates); until now €117 million
Germany	Gundremmingen-A	BWR, 250 MWe	11 years	Immediate dismantling – pilot project (underwater cutting)	~$300–550/kWe
Germany	Niederaichbach	Gas-cooled heavy water reactor, 100 MWe	1 year (1973–1974)	Immediate dismantling – pilot project	
Germany	17 reactors	VVER, PWR, HTR, BWR, GCHWR, 13–640 MWe	2–36 years (1961–2005)	Various – immediate to safe enclosure	
Italy	Caorso NPP	BWR, 840 MWe	3 years (1978 – closed in 1987 after referendum in 1986)	Safe enclosure: 30 years (internal demolition)	
Italy	Garigliano NPP (Caserta)	BWR, 150 MWe	18 years (1964–1982)	Safe enclosure: 30 years (internal demolition)	
Italy	Latina NPP (Foce Verde)	Magnox, 210 MWe gas-graphite	24 years (1962–1986 after referendum)	Safe enclosure: 30 years (internal demolition)	
Italy	Trino Vercellese NPP	PWR Westinghouse, 270 MWe	25 years (1962–1986 after referendum)	Safe enclosure: 30 years (internal demolition)	
Kazakhstan	Aktau BN-350	Fast neutron reactor, 52 MWe	27 years (1973–1999)	Safe enclosure	
Lithuania	Ignalina 1,2	RBMK LWGR, 408 MWe	28 years (1978–2008)	Safe enclosure	
Netherlands	Dodewaard NPP	BWR Westinghouse, 58 MWe	28 years (1969–1997)	Defueling completed – safe enclosure for 40 years	

(Continued)

Table 24.2 Continued

Country	Location	Reactor type	Operative life	Decommissioning phase	Dismantling cost
Russia	Six reactors	LWGR, BWR, VVER, 6–336 MWe	19–48 years (1954–2002)	Safe enclosure	
Slovakia	Bohunice A1	Gas-cooled heavy water reactor, 93 MWe	4 years (1974–1977)	Immediate dismantling	
Slovakia	Bohunice 1,2	VVER-440, 408 MWe	28 years (1978–2008)	Safe enclosure	
Sweden	Barseback 1,2	BWR, 600 MWe	24–28 years (1975–2005)	Safe enclosure	
Spain	Vandellós NPP-1	UNGG, 480 MWe (gas-graphite)	18 years *Incident*: fire in a turbogenerator (1971–1989)	Safe enclosure: 30 years (internal demolition)	Phases 1 and 2: €93 million
Spain	Jose Cabrera	PWR, 141 MWe	38 years (1968–2006)	Immediate dismantling	€135 million excluding spent fuel
Switzerland	DIORIT	MWe gas-graphite (experimental), 30 MWe	17 years (1960–1977)	Safe enclosure: years (internal demolition)	
Switzerland	LUCENS	8.3 MWe CO_2-heavy water (experimental)	1966–1969 *Incident*: fire in 1969	Entombment, safe enclosure and dismantling: 24 years (internal demolition)	
Switzerland	SAPHIR	0.01–0.1 MWe (light water pool)	36 years (1957–1993) (experimental demonstrator)	In public display since inauguration, open to visitors: 'Cherenkov's light'	
Ukraine	Chernobyl 1, 2, 3, 4	RBMK LWGR, 740–925 MWe	2–19 years (1979–2000) (fire and meltdown in Unit 4)	Safe enclosure	

Table 24.3 Nuclear decommissioning in Asia

Country	Location	Reactor type	Operative life	Decommissioning phase	Dismantling cost
China	Beijing (CIAE)	HWWR, 10 MWe (multipurpose) (heavy water experimental reactor for the production of plutonium and tritium)	49 years (1958–2007)	Safe enclosure and dismantling in 20 years (until 2027)	Proposed: \$6 million for dismantling, \$5 million for remote fuel handling
North Korea	Yongbyon	Magnox-type (reactor for the production of nuclear weapons through PUREX treatment)	20 years (1985–2005) Deactivated after a treaty	Safe enclosure: cooling tower dismantled	
Japan	Tokai-1	Magnox (GCR), 160 MWe	32 years (1966–1988)	Safe enclosure: 10 years, then dismantling until 2018	Estimated cost: Yen 93 billion (€660 million of 2003)
India	Tarapur-1, 2 (Maharashtra)	2 × BWR, 160 MWe	40 years (1969–2009?)	Not deactivated	
India	Rawatbhata Atomic Power Station-1, 2 (Rajasthan)	1 × PHWR 100 MWe, 1 × PHWR 200 MWe (similar to CANDU)	40 years (1970–2011?)	Not deactivated	
Iraq	Osiraq/Tammuz-1	BWR, 40 MWe (nuclear reactor with weapons-grade plutonium production capability)	Destroyed by Israeli Air Force in 1981	Not radioactive: never refurbished with uranium	?

These documents are supplemented by technical documents (TECDOCs) on specific topics which may be obtained from the IAEA website, www.IAEA.org.

24.9.2 Organization for Economic Co-operation and Development/Nuclear Energy Agency

The OECD/NEA has already been described earlier in the text for their major contributions to decommissioning technology and publications. Their continuing role in supporting and developing standards and guidance is a valuable resource for the industry. They may be accessed on the Internet at www.OECD-NEA.org.

24.9.3 United States of America: Office of Environmental Management – Department of Energy

In general, the US Department of Energy (US DOE) Office of Environmental Management (EM) is primarily concerned with decommissioning of former US weapons facilities. The US DOE/EM complex of facilities involves more than 150 sites in 30 states, and the decommissioning budget is currently estimated at $220 to $300 billion. The research and development work performed by the US DOE has direct relevance to NPP decommissioning, although no specific reference is made to NPPs. The US DOE has sponsored several decommissioning handbooks, the most recent one having been published by the American Society of Mechanical Engineers in 2004 (Toboas *et al.*, 2004).

As guidance for contractors performing decommissioning work within the weapons complex, the US DOE published DOE Orders describing the requirements for virtually all aspects of decommissioning and waste management. Additional information may be obtained from the US DOE website, www.em.doe.gov.

24.9.4 US Nuclear Regulatory Commission

The US Nuclear Regulatory Commission (US NRC) is the principal regulator for NPP decommissioning. Regulations for decommissioning are covered in Title 10 of the Code of Federal Regulations, Parts 2, 50 and 51, and there are related parts of the Code dealing with radiological health and safety, waste management and spent nuclear fuel. To provide guidance to licensees, the US NRC issued Regulatory Guides describing acceptable methods of complying with the regulations. Technical studies and reports on topical issues are published as NUREG documents, which provide additional non-binding guidance for licensees. The US NRC review staff prepared Standard

Review Plans identifying specific issues to be reviewed, and the acceptance criteria against which the decommissioning plans and procedures would be approved. The US NRC also published these documents for the licensees as further guidance. Additional information may be obtained from the US NRC website, www.nrc.gov.

24.9.5 United Kingdom

In the UK, the responsibility for managing the decommissioning of the commercial MAGNOX NPPs, uranium enrichment, fuel fabrication, fuel reprocessing facility, low-level waste disposal facility, and research and development facilities was assigned to the Nuclear Decommissioning Authority (NDA) in 2005. This quasi-governmental agency is responsible for funding all decommissioning work at these sites, a £73 billion liability. Some of the funding is derived from the remaining two operating NPPs, and the balance of the funding is provided by the Treasury. In 2006, the NDA issued its Strategy Document setting out the six principal goals of the NDA: Site Restoration, Business Optimization, Spent Fuels, Integrated Waste Management, Manage Nuclear Materials, and Managing the Critical Enablers. Additional information may be obtained from the NDA website, www.nda.gov.uk.

24.9.6 Other countries

Other countries have significant decommissioning programs underway, including Canada, France, Germany, Belgium, the Netherlands, Sweden, Spain, Italy, Switzerland, Austria, Slovenia, Russia, Lithuania, Romania, Bulgaria, Ukraine, Japan and Korea. It is not possible to cover all these countries in any depth in this chapter. Suffice it to say, the experience learned from recent decommissioning programs is being effectively applied in these countries. The Internet websites for these countries and their nuclear decommissioning programs can provide additional insight into their programs.

24.10 References

IAEA (1989), *Decontamination and Decommissioning of Nuclear Facilities,* IAEA TECDOC-511, International Atomic Energy Agency, Vienna.

IAEA (1999), *Decommissioning of Nuclear Power Plants and Research Reactors*, Safety Guide, Safety Standards Series, no. WS-G-2.1, IAEA, Vienna.

IAEA (2005a), *Financial Aspects of Decommissioning*, IAEA TECDOC-1476, IAEA, Vienna.

IAEA (2005b), *Selection of Decommissioning Strategies: Issues and Factors*, IAEA TECDOC-1478, IAEA, Vienna.

IAEA (2006a), *Decommissioning of Facilities Using Radioactive Materials,* Safety Requirements, Safety Standards Series, no. WS-R-5, IAEA, Vienna.

IAEA (2006b), *International Conference on Lessons Learned from the Decommissioning of Nuclear Facilities and the Safe Termination of Nuclear Activities*, Athens, Greece.

IAEA (2008), *Decommissioning of Nuclear Facilities: Training and Human Resources, Nuclear Energy Series* NG-T-2.3, IAEA, Vienna.

IAEA (2009), *General Safety Guide, Classification of Radioactive Wastes*, no. GS-G-1, IAEA, Vienna.

LaGuardia T S, *et al.* (1986), *Guidelines for Producing Nuclear Power Plant Decommissioning Cost Estimates*, AIF/NESP-036, Atomic Industrial Forum, Washington, DC.

Oak H D, *et al.* (June 1980), *Technology, Safety and Costs of Decommissioning a Reference Boiling Water Reactor Power Station,* NUREG/CR-0672 and addenda, Battelle Pacific Northwest Laboratory, for the Nuclear Regulatory Commission, Richland, WA.

OECD/NEA (1999), *Proposed Standardized List of Items for Costing Purposes in the Decommissioning of Nuclear Installations,* OECD/NEA, Paris.

OECD/NEA (2010), *Towards Greater Harmonization of Decommissioning Cost Estimates,* ISBN 978–92–64–99093–7, OECD/NEA, Paris.

Smith R I, Konzak G J, and Kennedy W E, Jr. (June 1978), *Technology, Safety and Costs of Decommissioning a Reference Pressurized Water Reactor Power Station,* NUREG/CR-0130 and addenda, Battelle Pacific Northwest Laboratory, for the Nuclear Regulatory Commission, Richland, WA.

Taboas A L, Moghissi A A, and LaGuardia T S (editors) (2004), *Decommissioning Handbook*, American Society of Mechanical Engineers, New York.

Tarcza G A (editor) (1987), *Proceedings of the 1987 International Decommissioning Symposium,* CONF-871018-Vol.1 (DE87012821), Westinghouse Hanford Company, Richland, WA.

US AEC (1968), *Termination of Operating Licenses for Nuclear Reactors*, Regulatory Guide 1.86, AEC, Washington, DC.

US Department of Labor (2010a), Bureau of Labor Statistics, *Employment Cost Index*, Series ID ecu13x02i (x being a variable for region of the country), Washington, DC.

US Department of Labor (2010b), Bureau of Labor Statistics, *Producer Price Index – Commodities*, Series ID wpu0543 [electricity] and wpu0573 [fuel oil], Washington, DC.

US NRC (2004), *Standard Review Plan for Decommissioning Cost Estimates for Nuclear Power Reactors*, NUREG-1713, Washington, DC.

US NRC (2005), *Report on Waste Disposal Charges: Changes in Decommissioning Waste Disposal Costs at Low-Level Waste Burial Facilities*, NUREG-1307, Revision 11, Washington, DC.

US NRC (2007), 'Method for Obtaining Approval of Proposed Disposal Procedures', *Title 10, Code of Federal Regulations, Part 20.2002,* Washington, DC.

US NRC (2011), 'Waste Classifications', *Title 10, Code of Federal Regulations, Part 61.55*, Washington, DC.

Appendix 1

The justification test for new nuclear power development: United Kingdom experience

W. E. A. WILSON, Burges Salmon LLP, UK

Abstract: Justification was first proposed as a regulatory principle by the ICRP, and in its original form it simply required that any practice involving radiation exposure should do more good than harm. This chapter considers the way in which the justification test has been developed as an IAEA Fundamental Safety Principle, elaborated by several key legal challenges and case law in the United Kingdom and the European Court of Justice, further developed in successive Euratom Directives, and is now being applied as a key component of regulatory consenting in the United Kingdom to the development of new nuclear power.

Key words: justification, Euratom, European Court of Justice and UK legal challenges, Justification Regulations.

A1.1 International Commission on Radiological Protection (ICRP) and origins

A1.1.1 Background

Justification was first proposed as a regulatory principle by the International Commission on Radiological Protection (ICRP), and in its original form it simply required that any practice involving radiation exposure should do more good than harm. As such, it has been a component part of radiological protection legislation for a number of years. The justification test has since been adopted by the International Atomic Energy Agency (IAEA) as one of its fundamental safety principles, and has been both elaborated by the ICRP and incorporated in a series of Basic Safety Standards Directives made under the Euratom Treaty.

This kind of test is relatively easy to apply where the advantages of a 'practice' are self-evident, notwithstanding its minor radiological detriments. For example, with smoke detectors the benefits are self-evident, while the detriments are reckoned to be small, despite the presence in some devices of americium-241. It is only when smoke detectors are stored in bulk, such as in warehouses, that regulation becomes an issue. The test is also straightforward to apply where the opposite is the case, and even a marginal radiological detriment is not really justified by the nature of the product. Examples frequently given are radioactive or luminous fishing floats. The Health Protection Agency at Harwell, England, has in a display stand at its training centre even stranger examples in this latter category, such as compressed gas cartridges for making carbonated drinks that proudly advertised the fact that they would add radon as well as carbon dioxide.

However, justification can be a harder principle to apply to more complex practices such as nuclear installations. In the United Kingdom between the early 1990s and 2001, the application of the justification test became greatly elaborated and complicated, in part in response to a series of legal challenges brought by anti-nuclear non-governmental organisations (NGOs) to government decisions, particularly over the THORP and MOX plants at Sellafield. This, together with brisk arguments within government departments about the precise application and scope of the justification test, led to very considerable elaboration of justification decisions, as government departments sought to head off challenges brought for judicial review of their administrative decisions by trying to show that each aspect of the test had been considered.

In the United Kingdom the justification test is now separated out from the rest of the Euratom Basic Safety Standards Directive implementation, and is applied by specially written regulations which describe how and by whom the test is to be applied. With the move towards the introduction of new nuclear power in the United Kingdom, application of the justification test will be an important milestone in the regulatory and consenting process, and may yet result in further legal challenges. The Nuclear Industry Association (NIA) has put forward several descriptions of design of new nuclear plant in an attempt to seek government approval of their justification as generic practices. The government has consulted publicly on these proposals, and a final determination was made in October 2010.

A1.1.2 ICRP 60

The ICRP is an independent international body of experts which provides guidance on topics related to protection of human health from the harmful affects of ionising radiation. ICRP Publication 60 provided that the detriment to be considered is not confined to that associated with radiation – it includes other detriments and costs of the practice. Often the radiation detriment will be a small part of the total (ICRP, 1991).

A1.1.3 ICRP 77

ICRP Publication 77 restated the justification test and made it clear that the justification of a practice requires only that the net benefit of the practice be positive. It stated that 'waste management and disposal operations are an integral part of the practice generating the waste. It is wrong to regard them as a free standing practice that needs its own justification' (ICRP, 1997, p. 13, para. 6.1.1 (34)).

In 2007 the ICRP made further recommendations that the occupation dose limit given in ICRP 60 in 1990 be retained, and that for planned exposure situations, during the normal operation of a nuclear power station, the limit should be expressed as 20 mSv per year, averaged over defined five-year periods, that is 100 mSv over five years without exceeding 50 MSv in any single year (ICRP, 2007, Executive Summary, pp. 11–16). This ICRP publication did not recommend any fundamental changes to the application of the main justification test.

Of course, the justification test advocated by the ICRP needs to be read in context with the other two tests recommended by that body, namely optimisation and dose limitation.

A1.1.4 IAEA Fundamental Safety Principles

The IAEA established a new primary IAEA Safety Standard, known as the Fundamental Safety Principles in September 2006. This brought together a broad international consensus on the elements of nuclear safety and protection against ionising radiation. It gave expression to 10 safety principles and explained their intent and purpose. These were to provide the grounds for establishing requirements and measures for the protection of people and the environment against the risks from ionising radiation, and to provide the basis for the safety of facilities and activities giving rise to such risks. One of the 10 principles was 'Justification of Facilities and Activities'.

The fundamental safety principle covering justification is quite simply set out in the IAEA Safety Standards Series. Principle 4 on justification of facilities and activities that give rise to radiation risks states that they must yield an overall benefit, and provides that 'for facilities and activities to be considered justified, the benefits that they yield must outweigh the radiation risks to which they give rise. For the purposes of assessing benefit and risk, all significant consequences of the operation of facilities and the conduct of activities have to be taken into account' (IAEA, 2006, para. 3.18).

The IAEA statement explains that decisions on benefit and risk are sometimes taken at the highest levels of government whilst in other cases it is appropriate for the regulatory body to determine whether the proposed facilities and activities are justified. Medical radiation exposure of patients is expressed to be a special case in that the benefit is primarily to the patient. The justification is considered specifically for each procedure and then on a patient-by-patient basis. Reliance is placed on the clinical judgement as to whether the procedure would be beneficial. As the IAEA notes, such clinical judgement is mainly a matter for medical practitioners who must therefore be properly trained in radiation protection. Beyond that, it may be noted that the IAEA expression of Fundamental Safety Principle 4 on justification is sufficiently wide that it would be hard to argue in the national context that any significant consequence of the operation of facilities and conduct of activities could properly be excluded from the consideration of justification.

A1.2 European Atomic Energy Community (Euratom) legislation and European Court of Justice and UK case law on justification

A1.2.1 Directive 80/836/Euratom

Justification as a principle of regulation of safety in the handling of ionising radiation was first adapted by the European Council (EC) in a Euratom Directive in 1980 (EC, 1980). Article 6(a) of Directive 80/836/Euratom provided that '. . . . every activity resulting in an exposure to ionising radiation shall be justified by the advantages which it produces. . . .' (p. 1).

A1.2.2 Directive 84/467/Euratom

Some minor changes were made to the incorporation of the justification test in Directive 84/467/Euratom (EC, 1984), where the relevant provision in Article 6(a) was changed to '. . . the various types of activity resulting in exposure to ionising radiation shall have been justified in advance by the advantages which they produce' (p. 4).

A1.2.3 Reading directives with ICRP standards

The important case before the European Court of Justice of Commission v Belgium (C-376/90) [1993] principally concerned minimum standards and the application in Belgian legislation of annual dose limits for adult workers. However, it is also read by a number of national authorities, including those in the UK, as authority for the proposition that the justification requirements set out in the Euratom Directives should be applied in the light of the latest recommendations of the ICRP, in that case in Publications 60, 75 and 77. The Court specifically acknowledged that Directives 80/836/Euratom and 84/467/Euratom were based on ICRP Publications; it noted (C-376/90, para. 23) the justification, optimisation and dose limit principles, and declared that 'it follows that the dose limits fixed by the ICRP are not absolute standards but are published only by way of guidance and that the principle underlying them is the optimisation of protection' (C-376/90, para. 25).

The importance of this case in the context of the UK government's approach to justification is that the UK government relies upon it as some authority for the proposition that each successive Euratom Basic Safety Standards Directive should be read in the context of progressive interpretations of the justification test in ICRP Recommendations. This is of particular significance in re-enforcing the UK government's view that taking European Council (EC) Directive 96/29/Euratom (EC, 1996) together with the latest ICRP Recommendations, the test required should be understood as a 'generic' one applied to a practice or class of practice, not a 'site specific' one applied, for example, to a particular installation in a particular place.

A1.2.4 The Greenpeace/Mr Justice Potts Decision 1994

This landmark UK case Regina v Secretary of State for the Environment, *ex parte* Greenpeace Ltd and Lancashire County Council in 1994 concerned the Sellafield Thermal Oxide Reprocessing Plant known as the THORP plant. In this case, the UK Court considered the 1980 Directive (as amended) and concluded amongst other things that Article 6(a) required a site-specific rather than a generic assessment. The UK government argued that the test had been met in substance – but the judge ruled that it had to be applied specifically. This was part of the origin of the very detailed and highly specific consideration which has been given to justification in subsequent decisions.

Mr Justice Potts dismissed the arguments of the UK government's advocate that Article 6(a) of Directive 80/836/Euratom required the justification in advance of a 'type of activity', not the carrying on of the activity at a particular site. The judge declared (Greenpeace/Mr Justice Potts Decision, para. 45) that 'I accept Mr Collins' [counsel for Greenpeace] submission that the principle of justification would be rendered meaningless if Mr Richards' [counsel for the UK government] construction was upheld. In my view ICRP 60 and the directive are concerned with justification of particular practices which affect particular individuals in particular circumstances. In this case the type of activity is thermal oxide reprocessing at Sellafield.'

The THORP judgement took place against a political background that was itself highly charged. THORP represented a £9 billion investment by the UK government, but there were 42,500 responses to one public consultation, many of them anti-nuclear. The finding that justification was a legal requirement applied on a site-specific basis was very significant.

A1.2.5 Revised Basic Safety Standards Directive 96/29/Euratom

Directive 96/29/Euratom (EURATOM, 1996, art. 6) revised the expression of the justification test case again. It provides that:

> ‘6.1 Member States shall ensure that all new classes or types of practice resulting in exposure to ionising radiation are justified in advance of being first adopted or first approved by their economic, social or other benefits in relation to any health detriment they may cause.
>
> 6.2 Existing classes or types of practice may be reviewed as to justification whenever new and important evidence about their efficacy or consequence is acquired.’

The UK government has interpreted this then and since as requiring a generic rather than a site-specific assessment of justification of ionising practices, citing its interpretation of Commission v Belgium as above.

A1.2.6 The Greenpeace/Mr Justice Collins Decision 2001

The decision of the Secretaries of State for Environment, Food and Rural Affairs (Margaret Beckett) and for Health (Alan Milburn) in the case of R (Friends of the Earth Ltd and Greenpeace Ltd) v Secretary of State for the Environment, Food and Rural Affairs, Secretary of State for Health on 3 October 2001 stated that:

> ‘We have concluded that the manufacture of MOX fuel is justified in accordance with the requirements of Article 6 of the Basic Safety Standards Directive 96/29/Euratom.’

But this followed extremely detailed consideration of the environmental, safety, economic, social and other benefits and disbenefits, carried out under the ever-present threat of judicial review, and even then Greenpeace challenged the treatment of ‘sunk costs’ in the construction of the Sellafield MOX plant. It took repeated rounds of consultation, economic reports from PA Consulting Group and Arthur D Little, draft decisions, reviews of new evidence such as data falsification incidents, defence of the challenges to the business case and so on before the decision could be concluded.

In the end, Mr Justice Collins held the MOX plant to have been justified, but held that construction and capital costs must be taken into account in future in deciding on economic benefits and detriments. The situation in the case whereby £300 million had already been spent on the plant before an application for justification was made to the Environment Agency was the subject of critical comment, which will need to be taken into account in future applications for new nuclear installations.

A1.2.7 Greenpeace challenge on nuclear policy 2007

Mr Justice Sullivan in the 2007 Greenpeace judicial review (The Greenpeace/Mr Justice Sullivan Decision) described the government’s consultation as ‘misleading’, ‘seriously flawed’ and ‘procedurally unfair’. The UK government’s Energy Review 2003 had described nuclear power as an 'unattractive option' with 'important issues unresolved' on waste. By 2006 the government stated that nuclear was 'expected to make a significant contribution to meeting our energy goals'. The judge held that 'something has gone seriously and radically wrong' with the consultation which was little more than an 'issues paper' and 'wholly insufficient for [consultees] to make an intelligent response' despite the fullest consultation having been promised.

All future consultations on any aspect of advancing the nuclear power agenda are bound to take note of the judgement. It represents an important marker put down by the courts that changes in policy in the nuclear field have to be properly explained and consulted upon, and if made too abruptly and without proper consultation are liable to be successfully challenged.

A1.3 UK regulations

A1.3.1 Ionising Radiations Regulations 1999

The Ionising Radiations Regulations 1999 implement the bulk of the 'radiation health and safety' provisions of Directive 96/29/Euratom, but after long debates within government it was concluded that topics such as justification went too wide to be incorporated in those regulations, and need to be treated separately.

This decision was taken partly because justification went beyond the exclusive jurisdiction and responsibilities of the Health and Safety Executive/Nuclear Installations Inspectorate, but also because of legal limitations to the enabling powers used as the basis for the 1999 Regulations. It was judged preferable to have a separate, stand-alone set of regulations to address the topic of justification.

A1.3.2 Justification of Practices Involving Ionising Radiation Regulations 2004

In the end, the Department for Business Enterprise and Regulatory Reform (BERR) in its *Justification of Practices Involving Ionising Radiation Regulations* established the process and procedure whereby the justification test is applied (BERR, 2004). There is a means of allocating a Justifying Authority to each such decision, in order to apply tests closely based on Article 6 of Directive 96/29/Euratom. It is probably fair to say that these regulations may have put an end to arguments before the UK courts on the procedure which should be followed in allocating a Ministerial decision maker to individual justification decisions, but they have by no means resolved all arguments about the scope and contents of the justification test.

These Regulations implement Articles 6(1) and 6(2) of Council Directive 96/29/Euratom and extend to the whole of the United Kingdom. They introduce the international radiological protection principle of generic 'justification' of classes of practices involving exposure to ionising radiation. They also implement certain prohibitions on the addition of radioactive materials to certain goods.

The Regulations prohibit practices unless they have been justified or were existing practices carried out before 13 May 2000, the date when the Directive came into force. They define the relevant Justifying Authority which takes the justification decisions, and this is the Scottish Ministers, The National Assembly for Wales or a Northern Ireland Department, to the extent allowed by devolution legislation within the United Kingdom, and for England and matters which are not devolved, the Secretary of State. The Regulations set out the conditions under which applications are to be made for justification decisions and allow conditions to be attached to those decisions, and further provide that the Justifying Authority may require operators or other persons to take appropriate steps in consequence upon a justification decision. The Secretary of State is permitted to decide whether a practice is new or existing.

The Regulations provide a mechanism for the determination of applications by the Justifying Authority, and the machinery whereby the Justifying Authority can require information from applicants of other persons and where necessary to hold inquiries or hearings and to carry out formal consultation with the public.

The Regulations give effect to certain bans on the addition of radioactive substances to personal ornaments, toys and cosmetics, and make separate provision for the justification of classes or types of practice involving medical exposure. In common with most regulations, provision is made for enforcement and offences whereby contravention notices can be served whether there are breaches of the Regulations and enforcement powers are given, with the usual suite of powers for the enforcement of criminal offences relating to breach of the Regulations' requirements.

A1.4 Application of justification test to nuclear new build proposals

A1.4.1 Nuclear Industry Association justification application

In May 2007 the Department of Trade and Industry (DTI) of the UK government published a public consultation on the role of nuclear power in a low-carbon UK economy and a technical consultation on a proposed process for the Regulatory Justification of new nuclear power stations (DTI, 2007).

In January 2008 the UK government published a White Paper on nuclear power (BERR, 2008, Annex B) which included its response to the technical consultation.

In March 2008 the government issued a call for Regulatory Justification applications for new nuclear power station designs, and the Department for Business Enterprise and Regulatory Reform (BERR) provided guidance on the level of information expected of applicants (BERR, 2008).

In November 2008 the trade association known as the Nuclear Industry Association (NIA) submitted on behalf of six utilities an application seeking justification by the Justifying Authority of four types of nuclear power stations, arguing that the benefits of their deployment would outweigh any radiological health detriment. The justification application submitted by the NIA was on behalf of six utilities, namely British Energy Group plc, EDF Energy plc, E.ON UK plc, RWE Npower plc, Vattenfall and Iberdrola Generacion S.A. It covered four specific reactor designs provided by the nuclear construction vendors, Atomic Energy of Canada Limited, Areva NPSAS, GE-Hitachi Nuclear Energy International LLC and Westinghouse Electric Company LLC.

The NIA noted that although the strict legal test set out in the Justification Regulations requires only that the benefits of a practice outweigh the radiological health detriments, the UK guidance on the process to be followed in applying the Basic Safety Standards Directive and the Justification Regulations to New Nuclear Power Stations takes this a stage further by 'suggesting' that it should be the net benefit that is weighed against the logical health detriment. The NIA states in the introduction to its application that this interpretation arguably goes beyond the Directive and the Regulations, although they state that they have followed this approach. The NIA notes further that the government's interpretation of the Justification Test means that it is necessary not to only assess the potential radiological health detriments associated with the practice, but also other potential detriments that could be significant when considered against the benefits derived from the practice. Accordingly, the NIA's application took a wide-ranging approach.

The application covers a discussion of the potential benefits that the practice could bring, especially focusing on the security of supply and climate change advantages of nuclear power, identification of the potential radiological health detriments, identification of other potential detriments associated with the practice, and finally a section comparing the net benefits and the radiological health detriments.

The NIA gave as the proposed practice to be submitted to the Justification Test:

> 'The generation of electricity from nuclear energy using oxide fuel of low enrichment in fissile content in light water cooled, water moderated thermal reactors using evolutionary designs.'

The NIA requested the Justifying Authority if it decided that the application comprised more than one class or type of practice to treat it as an application for justification of each such new class or type of practice. In the event, this is what the UK government has done by treating each reactor design as a separate practice or class of practices.

The NIA application went on to analyse the basic nuclear characteristics, design status and regulatory status for each of the designs, and the radiological health detriment under the headings of normal operation for workers, normal operation for the public and accident risk.

The benefits of nuclear power as claimed included security of supply and the beneficial effects on the carbon footprint. Extensive analysis was given to the issues of radioactive waste and decommissioning.

In the course of a lengthy application and argument, the NIA set out the industry's argument as to how radiological health detriments would be kept first of all within the mandatory exposure and dose limits required by legislation and secondly within levels that made them broadly comparable with one or two transatlantic flights.

In summary the NIA gave what it described as a high-level indicative assessment of the potential radiological health detriments associated with the development of new nuclear power stations, claiming that these would be extremely small and well within applicable regulatory dose limits, comparable to one additional return air flight from the UK to New York per year.

On radioactive waste and decommissioning the NIA maintained the industry line that in due course radioactive waste would be committed to a deep geological repository, and that secure arrangements were in place allowing for the safe storage of the nuclear waste likely to be generated by new nuclear power stations in the interim period.

Environmental effects included, for example, chemical effects from cooling water discharges, which were briefly dealt with and which in company with other forms of power generation are likely to be studied in greater depth and more critically assessed in the future. The risk of accidents from new nuclear power stations was considered and reckoned to be within bounds, controlled by legislation and properly addressed, manageable and small. Undoubted benefits of new nuclear power stations included security of supply benefits and carbon reduction benefits.

In summary, the NIA application claimed that the issues of radioactive waste, spent fuel and decommissioning had been properly addressed, wider environmental impacts were limited, and the economic assessment of new nuclear power was positive in line with government policy. There would be little change to existing small proliferation risks, the scale of potential radiological health effects was so low as to be of no concern, and overall the NIA concluded that the identified benefits for the UK from the proposed new practice are very significant while the detriments would be small and therefore the NIA sought that the proposed practice be justified.

A1.4.2 UK consultation on justification of new reactor designs, November 2009

On 30 October 2008, the UK government published a notice seeking further information, which was provided by the applicant trade association on 27 November 2008. The government then published a public consultation on the Secretary of State's proposed decisions as Justifying Authority on the regulatory justification of the new nuclear power station designs currently known as the AP1000 and the EPR in December 2008, and from that it became clear that the government had decided to treat each of the four proposed reactor designs as a separate class or type of practice, with a separate justification decision required for each. This consultation closed on 25 March 2009. Two of the reactor designs did not proceed at this stage, and the justification decision therefore went on to consider the AP1000 and EPR designs as separate applications.

In November 2009 the Secretary of State for Energy and Climate Change as the Justifying Authority issued proposed decisions in draft form for consultation on the NIA's Justification Application. The draft decision document published by the government goes on to consider in great detail issues such as radiological health detriment of nuclear power, issues of radioactive waste, environmental detriment, safety and security, carbon reduction benefits, security of supply benefits and an economic assessment.

The government's proposed decision document began by declaring that Regulatory Justification is a high-level, generic process confined to the relevant class or type of practice under consideration. It is not an assessment of government policy on whether to build new power stations, which was set out in the White Paper on Nuclear Power (BERR, 2008). Nor is it an exercise in comparing the advantages of the different methods of producing energy.

It will immediately be noted that wherever and whenever the government in its proposed decision document seeks to limit the scope of the decision being taken, there is an immediate tension with the full review of benefits and detriments that on some readings is required by the expression of the justification test, and in terms of judicial challenge, the more that is limited, the greater the risk that an objector can seek to show that the decision taker has overlooked a relevant consideration.

The document noted that a communication from the European Commission concerning the implementation of the Basic Safety Standards Directive states that compliance with the principle of justification 'can be safely assumed in respect of a new class or type of practice by the existence or laying down of regulations specifically concerning the class or type of practice'. Regulation 4(2) of the Justification of Practices Involving Ionising Radiation Regulation (BERR, 2004) defines 'justified' in relation to a new class or type of practice as 'justified by its economic, social or other benefits in relation to the health detriment it may cause'. The government document goes on to describe the other elements of tests applied to radiation control including optimisation and limitation, and the broader structure of regulation of the nuclear installations in the UK, including the generic design assessment applied to nuclear power station designs.

The Secretary of State as Justifying Authority took an early decision on a preliminary view that following the application submitted by the NIA, the justification decision should be by reference to four classes or types of practice based on each of the reactor designs submitted. In the event, those submitted by Atomic Energy of Canada Limited (the ACR1000) and GE-Hitachi of the USA (the ESBWR reactor) were not being taken forward within the generic design assessment applied by the United Kingdom, and therefore the justification decision being taken by the Secretary of State was applied to the two remaining reactor designs within the Generic Design Accreditation (GDA) process, namely the AP1000 designed by Westinghouse Electric Company LLC of the USA and the EPR designed by Areva NP of France and Germany.

A number of limiting assumptions are made in the course of the Secretary of State's lengthy proposed decision document of November 2009, which may yet be the subject of further legal challenge to the extent that they are reflected in the final decisions. These include the Secretary of State's conclusion that the Justification Authority is not bound to take practices outside the UK into account (see, for example, Vol. 2 Proposed decision on AP1000, paragraph 1.63). The recommendations of the ICRP require each country to assess the benefits and detriments of a class or type of practice carried on within its own borders and to enforce the conclusions from such assessments. The Secretary of State considers he has no authority to seek information from outside the UK. This is a curious conclusion, as surely if there is relevant information available from outside the UK it would be legally risky and practically mistaken to overlook it.

The government guidance for Regulatory Justification applications for new nuclear power stations (BERR, 2008) stated that applicants should provide information explaining how the proposed type or class of practice may cause radiological detriment to human health covering all aspects of the reactor lifecycle including, for example, decommissioning, waste disposal and transport.

The Secretary of State's proposed decision document then went through separate chapters on an analysis of radiological health detriment, radioactive waste, environmental detriment, safety and security, carbon reduction benefit, security of supply benefit, and economic assessment.

In Chapter 10 of Vol. 2 of the proposed decision on the AP1000, the Secretary of State sets out a draft decision that the class or type of practice is justified under the Justification of Practices Involving Ionising Radiation Regulations (BERR 2004), referring to the documents and evidence taken into account in Chapters 2 to 9 and the reasons for the proposed decision in Chapter 10. Under, for example, radiological health detriment, account is taken of the multiple containment systems and the practical measures and other regulations and powers used to ensure safety and security of reactors. Under radioactive waste the Secretary of State endorses geological disposal as the long-term answer for higher-activity waste, preceded by safe and secure interim storage. Environmental detriments are judged to be relatively limited, and safety and security considerations capable of being properly addressed. Security of supply and lower carbon benefits are judged to be considerable.

A1.4.3 The Secretary of State's decisions, October 2010

Finally, in October 2010, the Secretary of State announced separate decisions on the regulatory justification of the AP1000 and EPR nuclear reactor designs, which he found to be established. The justification decisions were given legal effect in the form of parallel sets of regulations, each with a substantial document setting out the background and reasoning. At the same time the Secretary of State published his decision that a public inquiry was not necessary, given the exhaustive consideration given over three years to the issues in the course of the review of regulatory justification.

That in summary is the way in which the UK government hopes to have concluded its consideration of justification, although as noted not every aspect of the Secretary of State's consideration is uncontroversial, and the Courts may still be invited in the course of a legal challenge to scrutinise the issue further.

A1.5 Conclusions

For about the last decade or so, anti-nuclear environmental NGOs have been very ready to bring legal challenges to the way in which government decisions applied the justification test, if any aspects, for example of economic assessment, had been left out of account. However, there is presently, at least for the time being, a certain amount of disarray within the environmental NGO movement on the subject of nuclear power, with several previously strong opponents having now declared in favour of nuclear power, arguing that it is the lesser of two evils when compared to climate change. This may make it slightly more unlikely that environmental NGOs will be quite as ready as they have been in the recent past to challenge justification decisions. However, they may still be minded to do so if those decisions disclose procedural mistakes or 'corner cutting' by government, for example if the UK government tries to combine too many forms of process into one single justification decision in order to save time, and thereby allows environmental NGOs to claim that more than 'practice' is being considered and justified at the same time.

It has been established by UK case law that the test to be applied is 'generic' rather than 'site-specific', in other words that it should apply to a whole practice rather than each power station on each site. The precise scope of each 'practice' has not yet been fully explored, and it is suggested that this is one area where, procedurally, the UK government could be at risk of further challenge if it was minded to compress the justification decision too much in order to achieve quick results. The result of a successful challenge by judicial review would be further delayed while the decision is re-taken, with full consideration being given to any aspect which has been left out of account.

Indications are that some NGOs, e.g. Greenpeace, are still solidly opposed to justification decisions. For example, a posting on the Greenpeace website on 12 January 2010 declared that 'It's difficult to find any, using the word of the UK government, "justification" for any of these designs. EPR comes out looking the worst of the four but only because it is actually off the drawing board and causing trouble in the real world. The other designs have as much potential for mayhem.'

NGOs have already noted strong links between this issue and the curtailment of opportunities to address wider concerns in the planning process, particularly by means of the Planning Act 2008 introduced by the last Labour government, so they may (and some evidently do) see justification as their last chance to force consideration of wider issues from uranium mining through to final waste disposal policies, risk of reactor accidents, terrorist attacks and handling of spent fuel.

Some will see this as ground for pressing for an inquiry, which is one option allowed for by the regulations (BERR, 2004, reg. 17) an option now ruled out by the Secretary of State's separate decision of October 2010.

For the UK, justification remains more than simply a technical framework for assessment of benefits and detriments, as it may have been originally envisaged by the ICRP. It is a test whose form has been influenced by the political debates and legal challenges that result from the fact that it is one of the important 'gateways' to the establishment of new nuclear power stations and other nuclear installations.

The Secretary of State's final decision on the justification of the two power station designs, as finally issued in October 2010, reflects both the regulatory origins and intentions of the justification test, and also a carefully prepared decision document that seeks to anticipate further legal challenges.

The events at Fukushima, Japan, in 2011 following the massive earthquake and tsunami, the well-reported difficulties with the nuclear reactor cooling systems, and the resulting radioactive contamination of land and sea will have been closely followed by environmental NGOs as well as governments and the nuclear industry around the world. In the United Kingdom the Secretary of State for Energy and Climate Change asked the Chief Nuclear Inspector for a report into the safety implications of the Fukushima disaster, and completion of the Generic Design Assessment of reactor designs was delayed until later in 2011 for this to be considered. Future applications of the justification test will be expected to show that relevant lessons have been learned, both in considering new justification applications under Article 6.1 of Directive 96/29/Euratom and when carrying out any review of existing classes or types of practice under Article 6.2.

A1.6 References

BERR (2004), *The Justification of Practices Involving Ionising Radiation Regulations 2004*, Statutory Instrument SI no. 2004/1769, Department for Business Enterprise and Regulatory Reform, London

BERR (2008), *Guidance for applications relating to new nuclear power*, URN 08/776, Department for Business Enterprise and Regulatory Reform, London

Commission v Belgium (C-376/90) [1993] 2 C.M.L.R. 513

DTI (2007), *The Future of Nuclear Power. The Role of Nuclear Power in a Low Carbon UK Economy, Consultation document on the proposed processes for justification and strategic siting assessment*, Department of Trade and Industry DTI/Pub 8547, URN07/972, London

EC (1980), Directive 80/836/EURATOM amending the Directives laying down the basic safety standards for the health protection of the general public and workers against the dangers of ionizing radiation, *Official Journal of the European Communities*, no. L 246, Office for Official Publications of the European Communities, Luxembourg

EC (1984), Directive 84/467/Euratom amending Directive 80/836/Euratom as regards the basic safety standards for the health protection of the general public and workers against the dangers of ionizing radiation, *Official Journal of the European Communities*, no. L 265/4, Office for Official Publications of the European Communities, Luxembourg

EC (1996), Directive 96/29/EURATOM laying down basic safety standards for the protection of the health of workers and the general public against the dangers arising from ionizing radiation, *Official Journal of the European Communities*, no. L 159, Office for Official Publications of the European Communities, Luxembourg

HSE (1999), *The Ionizing Radiation Regulations 1999*, Statutory Instrument no. 1999/3232. Health and Safety Executive, London

IAEA (2006), *Fundamental Safety Principles*, Safety Standards Series no. SF-1, IAEA, Vienna

ICRP (1991), The 1990 Recommendations of the International Commission on Radiological Protection, ICRP Publication 60, *Annals of the ICRP*, Volume 21, Issues 1–3, pp. 1–201. Pergamon Press, Oxford

ICRP (1997), Radiological Protection Policy for the Disposal of Radioactive Waste, ICRP Publication 77, *Annals of the ICRP*, Volume 27, Supplement 1, pp. 1–21. Pergamon Press, Oxford

ICRP (2007), The 2007 Recommendations of the International Commission on Radiological Protection, ICRP Publication 103, *Annals of the ICRP*, Volume 37, Issues 2–4, pp. 1–332. Elsevier

R (Friends of the Earth Ltd and Greenpeace Ltd) v Secretary of State for the Environment, Food and Rural Affairs, Secretary of State for Health [2001] EWHC Admin 914 (the Greenpeace/Mr Justice Collins Decision)

R (on the application of Greenpeace Ltd) v Secretary of State for Trade and Industry [2007] Env. LR 29 (the Greenpeace/Mr Justice Sullivan Decision)

R v Secretary of State for the Environment, *ex parte* Greenpeace Ltd and Lancashire County Council [1994] 3 C.M.L.R. 737 (the Greenpeace/Mr Justice Potts Decision)

Planning Act 2008

Public Inquiry Decision (DECC, October 2010)

Regulatory Justification decision on nuclear reactor AP1000 (DECC, October 2010)

Regulatory Justification decision on nuclear reactor EPR (DECC, October 2010)

The Justification Decision (Generation of Electricity by the AP1000 Nuclear Reactor) Regulations 2010 (Statutory Instrument number given when formally made)

The Justification Decision (Generation of Electricity by the EPR Nuclear Reactor) Regulations 2010 (Statutory Instrument number given when formally made)

The Justification of Practices Involving Ionising Radiation Regulations 2004. Consultation on the Secretary of State's Proposed Decisions as Justifying Authority on the Regulatory Justification of the New Nuclear Power Station Designs currently known as the AP1000 and the EPR. Vol. 1, Consultation Document; Vol. 2, Proposed decision on the AP1000 (DECC, November 2009)

Appendix 2

Nuclear safety culture: management, assessment and improvement of individual behaviour

A. CARNINO, Consultant in Safety, Management of Safety, Safety Culture and Security, France

Abstract: The nuclear industry has come to promote a safety culture in all its installations and activities. It is now recommended to develop a management system creating the environment necessary for individual and management behaviours and attitude fostering a good safety culture. In this Appendix are proposed ways of assessing its stage of development, ways for an organization to increase and manage it and ways for enhancing it towards excellence in safety.

Key words: safety culture, management, assessment, enhancement, individual behaviours, excellence in safety.

A2.1 Introduction

The nuclear industry has become aware of the importance of human factors and human errors through the accidents at Three Mile Island and Chernobyl as well as the accidents in the space industry. At the same time, operating experience has shown how human errors were a major factor in the events or near misses.

Following work on the man–machine interface, on the development of more adequate procedures, especially in accident conditions, and on the training of operators, the term 'safety culture' was introduced to indicate that a culture of safety would prevent most of the human errors and mistakes.

The term 'nuclear safety culture' was first used in the report of the International Nuclear Safety Advisory Group of the IAEA on the Chernobyl accident (INSAG, 1986). This pointed out the lack of safety culture. Another report of the same group followed on safety culture (INSAG, 1991), and gave examples of lack of it together with recommendations for enhancing it (INSAG, 2002).

Many other documents were later prepared for assessing safety culture, for improving it and for reviewing it, especially at the IAEA. A number of safety standards have now addressed this topic. The content of this

Published by Woodhead Publishing Limited, 2012

appendix relies on TECDOC-1329 of the IAEA, *Safety culture in nuclear installations* (IAEA, 2002), which put together the developments in safety culture in the previous 10 years.

A2.2 Definitions

The first definition was given by the INSAG-4 report on safety culture (INSAG, 1991): 'Safety culture is that assembly of characteristics and attitudes in organizations and individuals which establishes that, as an overriding priority, nuclear plant safety issues receive the attention warranted by their significance.'

The INSAG-4 report added attributes such as personal dedication, safety thinking and an inherently questioning attitude as intangible. Yet it is important to be able to judge the effectiveness of safety culture. INSAG has addressed it by starting from the perception that the intangible attributes lead naturally to tangible manifestations that can act as indicators of safety culture.

Another definition is given and used by the Institute of Nuclear Power Operations (INPO, 2004): 'Safety culture: an organization's values and behaviours – modelled by its leaders and internalized by its members – that serve to make nuclear safety the overriding priority'. INPO then establishes the following safety principles:

- Personal responsibility for nuclear safety
- Leadership commitment to safety
- Trust permeates the organization
- Decision-making reflects safety first
- Recognize unique nature of nuclear
- Cultivation of questioning attitude
- Embracement of organizational learning
- Constant examination of nuclear safety.

These definitions and considerations lead then to the safety principle which is part of the new safety fundamentals of the IAEA (IAEA, 2006a):

> 'Principle 3: Leadership and management for safety
>
> Effective leadership and management for safety must be established and sustained in organizations concerned with, and facilities and activities that give rise to, radiation risks.'

During the development of the International Nuclear Safety Convention, it was pointed out that a culture is a diffuse matter and therefore cannot be referred to in a legal context. Safety culture was therefore transformed into 'Safety First'.

From these definitions and considerations, one should remember the two major components: the organization and the individual behaviour.

A2.3 The organization

A2.3.1 Management system

The IAEA Safety Requirements entitled *The Management System for Facilities and Activities* (IAEA, 2006b) indicate that:

> 'A management system shall be established, implemented, assessed and continually improved. It shall be aligned with the goals of the organization and shall contribute to their achievement. The main aim of the management system shall be to achieve and enhance safety by:
>
> – Bringing together in a coherent manner all the requirements for managing the organization;
> – Describing the planned and systematic actions necessary to provide adequate confidence that all these requirements are satisfied;
> – Ensuring that health, environmental, security, quality and economic requirements are not considered separately from safety requirements, to help preclude their possible negative impact on safety.
>
> Safety shall be paramount within the management system, overriding all other demands.'

Within the organization, the Safety Requirements cited above add that the management system shall be used to promote and support a strong safety culture by:

- Ensuring a common understanding of the key aspects of safety culture within the organization
- Providing the means by which the organization supports individuals and teams in carrying out their tasks safely and successfully, taking into account the interaction between individuals, technology and the organization
- Reinforcing a learning and questioning attitude at all levels of the organization
- Providing the means by which the organization continually seeks to develop and improve its safety culture
- Taking into account the complexities of processes and their interactions.

This means that the organization has to advertise to all staff its objectives in terms of safety, to obtain from all managers within its structure the support needed to create the right environment that will induce the needed attitudes and behaviours:

- Individual and collective commitment to safety on the part of the leadership, the management and personnel at all levels
- Accountability of organizations and of individuals at all levels for safety
- Measures to encourage a questioning and learning attitude and to discourage complacency with regard to safety.

The difficulty in the management system is to identify all interactions between individuals, including where appropriate the interactions with personnel from contractors: for example, the situation during outages for refuelling or maintenance in a power plant.

For ensuring a good safety culture, the management should regularly assess its safety culture developments, listen to its staff, learn from operating experience and where appropriate promote actions for the promotion of safety culture. Learning from human errors or organizational failures should also be a means of improving safety culture.

A2.3.2 Stages of development of safety culture

For evaluating the situation of the organization in terms of safety culture, it is possible to identify its stage of development. The three main stages in the development of safety culture have been defined in the IAEA technical document on safety culture in nuclear power plants (IAEA, 2002) as follows.

Stage 1: Safety is based on rules and regulations

- Problems are not anticipated, and the organization reacts to each one as it occurs.
- Communications between departments and functions is poor.
- Collaboration and shared decision-making is limited.
- People who make mistakes are blamed for their failure to comply with the rules.
- The role of management is seen as enforcing the rules.
- There is not much listening or learning inside or outside the organization, which generally adopts a defensive position when criticized.
- People are viewed as components of the system – the mechanistic view.
- There is an adversarial relationship between managers and other employees.
- People are rewarded for obedience and results, regardless of long-term consequences.

At this stage, safety is perceived as a constraint very often imposed by the regulator, and human errors are viewed as a problem requiring sanctions. Fulfilling all safety rules is the prime objective. The management does not pay much attention to behaviours and attitudes. A regulator who is very prescriptive creates such a perception of safety and thus influences safety culture. The organization is not open to external exchanges and could even be very isolated.

Stage 2: Safety is considered an organizational goal

- There is growing awareness of the impact of cultural issues in the workplace, although it is not understood why added controls and training have not yielded the expected safety improvements.
- Management encourages interdepartmental and interfunctional communications.
- Management's response to mistakes is to introduce more controls and procedures and to provide more retraining.
- The role of management is to make sure that goals are achieved and that work objectives are clear to employees.
- The organization is willing to learn from external groups, especially new techniques and best practices.
- The relationship between employees and management is adversarial, although there may be more opportunities to discuss common goals.
- People are rewarded for exceeding goals regardless of long-term consequences.
- The interaction of people and technology is considered, but more from the viewpoint of increasing the efficiency of the technology.
- There is more teamwork.
- The organization remains reactive in relation to problems, although there may be more anticipation of potential problems in planning.

The organization in stage 2 has become aware of the importance of safety, knowing that good safety goes with good availability. Communication starts to be open and external exchanges are perceived fruitful. Managers are present in the shops but it is not yet a blame-free environment. Safety performances are the main objective.

Stage 3: Safety can always be improved

- Problems are anticipated and dealt with before they occur.
- Collaboration between departments and functions is good.
- There is no goal conflict between safety and production.
- Almost all mistakes are viewed in terms of process variability with the emphasis placed on understanding what has happened, rather than finding someone to blame.
- Management's role is seen as coaching people to improve performance.

- Learning from others, both inside and outside the organization, is valued.
- People are respected and valued for their contribution.
- The relationship between management and employees is mutually supportive.
- People are aware of the impact of cultural issues, and these are considered in decision making.
- People are rewarded for improving processes, as well as results.
- People are considered to be an important part of organizational systems with attention given to satisfying their needs, and not just to achieve technical efficiency.

An organization in stage 3 has adopted the idea of continuous improvement and applied the concept to safety: it is a learning organization. There is a strong emphasis on communications, training, management style and improving efficiency and effectiveness. People within the organization understand the impact of cultural issues on safety.

The time-scale required to pass through the various stages cannot be predicted. Much will depend upon the circumstances of an individual organization, and the commitment and effort that it is prepared to make in order to bring about change. Sufficient time must be taken at each stage to allow the benefits from changed practices to be realized and to mature. It should be remembered that an organization might possess characteristics associated with each of the three stages. Change in an organization is rarely simultaneous or uniform. A rule-based approach should not be viewed negatively. There will be activities or circumstances in organizational life where strict compliance with rules is essential, e.g. emergency response, or operating with sufficient margin for safety. Cultural awareness is not incompatible with having strict rules; much of culture is about complying or conforming to norms.

A2.4 Assessing the stage of development of safety culture

The stage of development of safety culture in an organization can be assessed using a simple method. Individuals can use the method separately or in groups. The method is based on how the organization being assessed views certain factors such as mistakes, time, role of managers, handling of conflict and the nature of people. Each of these factors is viewed in a slightly different way in each of the three stages of development of safety culture.

The approach is to consider which stage is most reflective of the factor being considered. The IAEA document on safety culture in nuclear power plants (IAEA, 2002) describes the methodology to be used in assessing the status of safety culture in a given plant. Depending on the stage characteristics, one can summarize how these factors are viewed in Table A2.1.

Another technique for assessing the safety culture can be used, which is an employee survey. The survey can be performed by interviews conducted orally or via a written questionnaire. Such surveys may have an important cost and the benefits have to be balanced against that cost. This first needs an explanation of the objectives pursued and the designation of a team of specialists including statisticians and work psychologists for collecting the information and carrying the interviews. Second, installation staff should be

Table A2.1 Ways in which safety culture factors are viewed at different stages of development of safety culture

View of mistakes	
Rule-based	People are blamed for non-compliance with rules. Organizations react defensively to criticism rather than listening and learning.
Goal-based	Mistakes result in more controls and training.
Improvement-based	Mistakes are an opportunity to understand and improve.
Time focus	
Rule-based	Short-term is all-important.
Goal-based	People are rewarded for exceeding goals, regardless of long-term consequences. Numerical targets are specified for safety.
Improvement-based	Short-term performance is analysed to improve longer-term performance. Longer-term focus with anticipation of consequences.
Role of managers	
Rule-based	Managers enforce rules and pressure employees for results.
Goal-based	Managers use techniques such as management by objectives.
Improvement-based	Managers coach people to improve performance. Managers support collaborative work.
Handling of conflict	
Rule-based	Conflicts are rarely resolved and groups continue to compete with one another.
Goal-based	Conflict is discouraged in the name of teamwork.
Improvement-based	Conflict is resolved by means of mutually beneficial solutions.
View of people	
Rule-based	People are components in a system.
Goal-based	Growing awareness that people's attitudes influence their performance.
Improvement-based	People are respected and valued for their contribution.

convinced that the results will be used to improve the work environment and the management style. They expect to be informed of all results and of the actions envisaged as a result. They will in this way have a certain appropriation of the progress made.

To be significant, the interpretation of the survey needs careful analysis: for example, the lack of an answer to some questions has to be interpreted and may lead to additional interviews or questions. The statistical analysis is also difficult if the sample of personnel answers is limited.

A2.5 Identifying the lack of safety culture

From the operating experience and the analysis of events showing a declining safety culture in organizations, it was possible to list the symptoms which were pre-existing and to which the organizations did not pay attention. The IAEA document on safety culture in nuclear power plants (IAEA, 2002) also gives the symptoms of a lack of safety culture. These are:

- Lack of systematic approach to safety
- Procedures not properly serviced
- Incidents not analysed in depth and lessons not learned
- Resource mismatch
- Violations increasing in number
- Increasing backlog of corrective actions
- Verification of readiness for operation or maintenance
- Employee safety concerns not dealt with promptly
- Disproportionate focus on technical issues
- Lack of self-assessment processes
- Poor housekeeping
- Failure of corporate memory
- Low status of Quality Assurance department
- Lack of corporate oversight
- Lack of ownership
- Isolationism
- Lack of learning
- Unwillingness to share or cooperate
- Failure to deal with the findings of independent external safety reviews
- Deficiencies in regulatory bodies.

Monitoring of safety culture has become a must. Indicators are not easy to define but should be determined when actions for improving safety culture are taken and evaluated. Again it is only the tangible manifestations which can be observed.

Learning from event analysis, five major causes of declining safety culture were identified:

- Over-confidence: good past performance leading to self-satisfaction.
- Complacency: occurrence of minor events that are subjected to minimum self-assessment, and delay in improvement programmes.
- Denial: number of minor events increases, with possibly a more significant event. These are treated as isolated events. Findings from audits are considered invalid. Root cause analysis is not used.
- Danger: several potentially serious events occur but management and employees reject criticism from audits or regulator, by considering their views biased. The oversight function is afraid to confront management.
- Collapse: regulator intervenes to implement special evaluations. Management is overwhelmed and may need to be replaced. Major and very costly improvement needs to be implemented.

A2.6 Improvement of safety culture

A2.6.1 Senior management

Beyond the organization of management of safety culture, senior management has to support the measures taken and themselves to actively participate in creating the work environment which will allow stage 2 at least and give the trends to stage 3 of safety culture development. As indicated in the IAEA document on safety culture in nuclear power plants (IAEA, 2002) senior managers should:

- Gain an understanding of safety culture as a concept
- Be visibly interested in safety and integrate it into their other activities
- Encourage employees to have a questioning attitude on safety
- Ensure that safety is included in planning activities
- Regularly review safety to ensure its adequacy for current and future circumstances
- Monitor safety trends to ensure that safety objectives and performances are being achieved
- Recognize those who improve safety.

In addition they should also ensure prevention of accidents, which is one of the objectives of safety. To achieve it, risk analysis is performed, together with a thorough analysis of events, errors should be seen as learning opportunities, and the organization itself is also learning. Employees should be seen as contributors to the safety improvements. Contractors need to be consulted and to have a common understanding of safety culture. An essential means of having the concept of safety culture fully understood and shared between all is training and retraining, since a single injection of safety culture is not sufficient to maintain it throughout the lifetime of an installation.

Senior management, having to permanently monitor the level of safety culture, can rely on self-assessment of safety culture or organize peer reviews either internally or with external experts. Safety performance may give information on the achievement in terms of safety culture. Although difficult to outline, safety indicators linked to safety culture might help. Examples of such indicators could be:

- Percentage of safety improvement proposals implemented during the previous month or quarter
- Number of safety inspections conducted by senior managers during the past month
- Number of employees who have received refresher safety training during the past month
- Number of safety audit recommendations implemented during the past month.

A2.6.2 Regulatory body

The regulator has a strong influence on the development of safety culture in the organizations under its jurisdiction. If the safety approach is rule based, it will influence the organizations towards stage 1 of development of safety culture. If the approach is aiming at monitoring the performances and following given indicators, safety culture will develop towards stage 2. But if the regulator concentrates on the organizational management system, the trend in safety culture will be a learning organization much more tuned to always improving safety.

Regular exchanges of information on the safety culture improvements and trends need to take place between the regulator and the organization with respect of the respective responsibilities. Training of all regulatory staff on safety culture is necessary to fully understand its evolution in the organization. Especially, the inspectors need to internalize the safety culture concept.

Having an external view, the regulator may notice some of the signs of decline of safety culture and ask the organization to take relevant actions. Examples of such signs could be loss of corporate memory, low quality assurance, role of headquarters in financial and human resources, lack of commitment of top management to safety, and lack of openness in communication and sharing with others.

A2.6.3 Individuals

All the staff of the organization must be imbued with the concept of safety culture and trained on it. Their attitude and behaviour should reflect the search for excellence and their commitment to safety in all circumstances. It is not a new constraint, it is their personnel conviction that the organizational environment is facilitating the work required and that the management works to motivate, listen and value the individuals and teamwork.

Some behaviour is expected from all:

- A questioning attitude which leads one to ask oneself questions like:
  - Do I know the work I have to perform? Do I have the right equipment/materials?
  - If something goes wrong what am I supposed to do?
  - What is the importance of my work for safety?
  - What could go wrong and what would be the consequences on safety?
  - Is it teamwork? Is the team aware of its responsibilities?
  - Do I have the right procedure?

Many other questions can come to mind for which the answers are generally obvious. An important one is: when facing an abnormal condition or environment to whom should I report?

- A prudent and rigorous approach, so important for new tasks or modified ones:
  - Implementation and understanding of procedures and when possible participation in the elaboration of new ones.
  - If something goes wrong, stop, think, act and report.
  - Ask for assistance when necessary.
  - Do not forget the responsibilities given to you or to the team you belong to.
  - Accept being part of a learning organization.
- Communication
  - Being part of an organization setting safety improvements as a goal requires good communication vertically (both up and down) and transversely.
  - Suggest safety improvements when needed based on your professionalism and your experience.
  - Promote safety culture in your team.
  - Be aware of the importance of your tasks *vis-à-vis* global safety and communicate to your colleagues.
  - Obtain useful information from others.
  - Transmit information to others.
  - Report on the results of your work, whether it is usual or new.

Means for implementing a continuously improving safety culture need to be considered for all personnel and individuals, who should be consulted for creating the environment which they need for exercising their professionalism and their skills within the organization. Their opinion is of great importance in areas such as setting the workplace and adapting the workload, giving the right indications on what is happening in the installation, and putting in place mechanisms for communicating to supervisors and mechanisms for rewarding safety improvements. The regular presence of managers in the workplace increases motivation and gives a role model to follow.

A2.7 Conclusion

If a lot of people think that safety culture is bringing a new set of obligations and rules. Most of the time, this is wrong. A normal reaction of individuals is to resist change. Certainly resistance to change is due to not understanding the safety culture concept. The culture cannot be changed overnight and the approach should be gradual. All plant personnel will finally benefit from the actions taken through their involvement, through the improvements in workplaces as well as in the global management initiatives of learning and listening to all. Professionalism, skills and communication at all levels are recognized and participate in the development of stage 3 of safety culture. Creativity, including openness to the creativity of others, is critical to the success of a change programme and requires the leader's openness to considering and trying new ideas. Appropriation of the job's changes, learning at all times and communicating are keys to the success of safety culture.

A2.8 References and further reading

IAEA (2002), *Safety Culture in Nuclear Installations,* TECDOC-1329, IAEA, Vienna

IAEA (2006a), *Fundamental Safety Principles,* Safety Standards Series, Safety Fundamentals SF-I, IAEA, Vienna

IAEA (2006b), *The Management System for Facilities and Activities,* Safety Standards Series, Safety Requirements GS-R-3, IAEA, Vienna

IAEA (2010), *International Standards, Codes and Guides,* IAEA, Vienna

INPO (2004), *Principles for a Strong Nuclear Safety Coltore*, Institute of Nuclear Power Operations, Atlanta, GA

INSAG (1986), *Summary Report on the Post-Accident Review on the Chernobyl Accident,* Safety Series no. 75-INSAG-l, IAEA, Vienna

INSAG (1991), *Safety Culture,* Safety Series no. INSAG-4, IAEA, Vienna

INSAG (1992), *The Cherbobyl Accident: Updating of INSAG-I,* Safety Series no. INSAG-7, IAEA, Vienna

INSAG (2002), *Key Practical Issues in Strengthening Safety Culture,* Safety Series no. INSAG-15, IAEA, Vienna

Appendix 3
Nuclear installation safety: International Atomic Energy Agency (IAEA) training programmes, materials and resources

M. J. MORACHO RAMIREZ, International Atomic Energy Agency (IAEA), Austria

Abstract: This chapter discusses training materials, resources and programmes provided by the International Atomic Energy Agency (IAEA). The chapter reviews web-based training, regional cooperation and harmonised training programmes as well as training based on TECDOC-1234 and SARCoN guidelines.

Key words: International Atomic Energy Agency (IAEA), web-based training, harmonised training programmes, TECDOC-1234 and SARCoN guidelines.

A3.1 Background and introduction

The IAEA Fundamental Safety Principles (IAEA, 2006a), principles 2 and 3 in particular, underline the need and importance of having technical and managerial competence as well as having in place appropriate management systems to ensure these resources. Human resource development (HRD) is an essential requisite for the safety and sustainability of a nuclear power plant (NPP). To build and maintain a competent workforce is particularly complex for countries embarking on a nuclear power plant programme. Guidance documents from the Agency underline the importance of ensuring the necessary competence in safety. The IAEA Safety Guide SSG-16,

Published by Woodhead Publishing Limited, 2012

Establishing the Safety Infrastructure for a Nuclear Power Programme, to be published in 2011 additional information can be obtained from: http://www-ns.iaea.org/tech-areas/safety-infrastructure/default.htm, identifies actions that should be realized by the government, regulatory body, operating organization and other relevant organizations in the initial three phases of a NPP (i.e. before the decision to build the first nuclear power plant until the plant is commissioned to operate (IAEA, 2008; INSAG, 2008)). In the INSAG-22 report (INSAG, 2008), consistent with IAEA (2008), the lifetime of a nuclear power plant is divided into five phases from a nuclear safety standpoint and indicative average durations are provided for each of these phases.

- Phase 1 is 'Safety infrastructure before deciding to launch a nuclear power programme' (average duration: 1–3 years).
- Phase 2 is 'Safety infrastructure preparatory work for construction of a nuclear power plant after a policy decision has been taken' (average duration: 3–7 years).
- Phase 3 is 'Safety infrastructure during implementation of the first nuclear power plant' (average duration: 7–10 years).
- Phase 4 is 'Safety infrastructure during the operation phase of a nuclear power plant' (average duration: 40–60 years).
- Phase 5 is 'Safety infrastructure during the decommissioning and waste management phases of a nuclear power plant' (average duration: 20 to more than 100 years).

The present appendix uses the same approach in considering phases 1, 2 and 3.

A video presentation introducing the purpose and content of the IAEA safety guide can be found at http://www-ns.iaea.org/downloads/video/ni/training-ds424/index.htm.

Nuclear technology, based on nuclear physics and other advanced sciences and technologies, demands high levels of knowledge and experience as priority must be given to safety. Decision makers must be aware that building nuclear safety competence is a multidisciplinary and multi-institutional undertaking with a scope, level of effort and cost well beyond that normally required for other industrial developments. This awareness is essential for an informed national commitment, if a decision to embark on a NPP is made. It is essential to build institutional knowledge, and to attract, train and sustain a competent workforce capable of conducting a safe and reliable nuclear programme. This includes managerial and subject knowledge with a special focus on nuclear safety matters. The following aspects should be considered:

- Evolving needs in the various phases of safety infrastructure
- Target persons and organizations to receive training

- Type of knowledge required
- Depth of knowledge required.

Depending on the phase of development and resources of the country, these aspects may vary. When a decision on whether or not to embark on nuclear power is to be made as in phase 1, the type and depth of knowledge, target people and organizations are different than in phase 3 when the type of plant and reactor design are already known. Also the number of personnel needed and the time-frame for training the workforce are important parameters to take into account in planning to build the necessary competence.

Related to the main aspects of the safety infrastructure and actions and actors involved in the application of the safety standards, the IAEA has identified 11 modules for which safety packages have been produced, including the related safety standards, available tutorial materials, review services, and workshops and seminars:

- Module 1 – Governmental, Legal and Regulatory Framework for Safety
- Module 2 – Human Resources Development
- Module 3 – Leadership and Management for Safety
- Module 4 – Radiation Protection
- Module 5 – Site Survey, Site Selection and Site Evaluation
- Module 6 – Safety of Radioactive Waste, Spent Fuel and Decommissioning
- Module 7 – Emergency Preparedness and Response
- Module 8 – External Support Organizations and Contractors
- Module 9 – Safety in Design, Safety Assessment and Research for Safety
- Module 10 – Transport Safety
- Module 11 – Interaction with Nuclear Security

A3.2 Building competence and effectiveness of training

The need for and importance of nuclear competence are underlined in the IAEA safety standards as they have an impact on safety (IAEA, 2002, 2006b,c, 2010). In building the necessary capacity and competence for a country, one must emphasize the convenience of using the national resources: universities, research centres and industrial institutions, up to the maximum possible level, in those training activities. Various dimensions should be considered: institutional, technical, organizational and managerial. Training must be tailored to the different areas and target audiences. For example, a training programme for enhancing the institutional capacity might be focused on the national structure and study of successful institutions in other countries; a training programme for improving the technical areas might use courses from universities and training materials from the industry; and a training programme for effective management can benefit from several successful experiences in the nuclear field. Given the comprehensive lists of subjects and training materials, there is a risk for the embarking country to try to fulfil a variety of courses without having conducted proper analysis and identification of needs, target audience, phase of development of the country, and analysis of resources available at a national regional and international level. However, the success of training lies in a good system for management of competent and well-trained trainers.

Effective training goes beyond the actual training materials and courses. It must be framed in a management system including training needs assessment, design of the training for fulfilling the identified needs through conscientious identification of learning objectives, time allocation, methods of training, exercises, appropriate materials, implementation through interactive and motivational techniques, and assessment of the training. One of the prime training target groups is the high-level management of the project who have an important role in defining the needs for specific training along the life of the project. For instance, there should be well-defined training programmes for technology selection, site analysis, design and construction, quality assurance, commissioning and operation, all under the ownership of the managers of such activities. This ownership is of particular importance during the operation phase. Training managers must have the necessary expertise to analyse needs, and these needs should be defined in close collaboration with the managers of the activities to be analysed. Moreover, they must assign adequate care to the design and implementation of the training to fulfil the specific gaps. This last aspect does not always receive the necessary attention as we tend to believe that the best experts are the best trainers disregarding the fact that they might fail to communicate, involve, motivate, respect the trainees and follow their development. 'Train the trainers' programmes are one of the most efficient ways of investing in training, if effectiveness is searched for. Aspects of training such as communication must also take account of the national culture.

The IAEA has developed comprehensive safety competence frameworks for identifying training needs, and training materials and curricula for those competence areas. It also provides support through seminars for training managers that focus on the use of the IAEA training materials and documents for designing tailored programmes for the training needs of the organization. Practical training is supported through fellowships.

Mention must be made here of the IAEA Systematic Assessment of Regulatory Competence Needs (SARCoN) guidelines (http://www-ns.iaea.org/training/ni/tools-networking.asp?s=9&l=75) which is a revision of TECDOC-1254, *Training the staff of the regulatory body for nuclear facilities: A competency framework* (IAEA, 2001). These guidelines explain the process of systematic training needs assessment and include questionnaires for self-assessment of competence needs with more than 100 competencies under a four-quadrants framework. The Annex in Section A3.13 gives examples of these competencies and how they are organised along four quadrants.

A3.3 Training of leaders for safety, emerging regulators

Training of leaders for safety is a determining component of an effective strategy for building competent organizations. The question often arises of what it takes to build a competent regulatory body in a country that wants to embark on nuclear power. In answering this question, some struggle, giving advice on the number of staff and qualifications of all staff, technical areas, costs and time frames. There are no prescribed models of regulators recommended in the IAEA safety standards. Reality shows that depending on the regulatory approach, culture and infrastructure of the country, the regulator's staff numbers vary dramatically. The number of staff needed depends heavily on the degree of outsourcing of competence at a national, regional or international level. Also the national system of technical safety and support organizations is an important variable in the equation. An organization of 50 perfectly qualified technical people does not necessarily constitute a regulator. The regulator is defined within a system of people and processes resulting from national and international experience and cooperation, knowledge exchange and cultural factors. The most effective approach for building a competent nuclear safety organization is not only training a number of people in technical areas (i.e. see quadrant 2, Fig. A3.1) but also designing and implementing training for leaders in safety values (i.e. see along the lines of quadrant 4, Fig. A3.1). Training a reduced group of effective leaders and managers might be the best way of achieving the optimal resources. Excellent safety leaders would learn the right values for safety, draw wisdom from the international environment, adapt the knowledge in their cultural environment, find the tools, seek the necessary agreements, make efficient use of national resources, and cooperate regionally and internationally to acquire and maintain the necessary competence. Ten well-trained leaders and managers who take responsibility and commitment might be more efficient than 50 trained engineers. Training for leaders and managers to use efficiently all the international knowledge, information and resources already available is a real challenge in the training programmes.

A3.4 Challenges for building sustainable competence systems

Due to the limited resources of the IAEA for the increasing demand for training that stems from the interest in safe nuclear development, the

1. Legal basis and regulatory process competencies · Legal basis · Regulatory processes · Regulatory guidance documents · Licence and licensing documents · Enforcement process	**2. Technical disciplines competencies** · Basic technology · Applied technology · Specialized technology
3. Regulatory practices competencies · Safety-focused analytical techniques · Inspection techniques · Assessment techniques · Investigation techniques	**4. Personal and interpersonal effectiveness competencies** · Analytical thinking, problem solving and decision making · Personal effectiveness · Communication · Teamwork · Management

A3.1 Four quadrants competencies model based on TECDOC-1254 and SARCoN guidelines.

question often arises of what would be the most effective strategy for countries to build the necessary safety competence in a sustainable way. It is important to seek the best basis for absorbing external support in a sustainable way. Factors that play an important role and need to be optimized are the national capacity and infrastructure of the country and the resources from regional and international cooperation. At a national level it is necessary to analyse the needs at an institutional, organizational, managerial and technical level. This analysis will help to identify measures to strengthen the 'national capacity for building competence in a sustainable way'.

There is a need for optimization of national capacity, regional and international resources. It is important to underline knowledge transfer and ownership. There is a proliferation of offers for international courses in nuclear safety and this might make it difficult to be an intelligent customer and choose the best outsource of training for the needs of the organization, operator or institutions of the country. Expensive contracts for training can be signed that do not always result in effective transfer of knowledge. The trainees benefit individually but do not necessarily use and transfer that knowledge to others in their home country. It is recommended that any outsourcing of knowledge and training includes in its agreement or contract a clause for knowledge transfer. Training received externally or outsourced should include as an additional activity that trainees be instructed as trainers in order to enable them to design a repetition of the received training back in their organization through adapting it to their needs. Organizations should have adequate systems to capture the knowledge, retain it and repeat the training in-house.

A3.5 IAEA training materials and related resources

The IAEA developed a strategy for its activities related to education and training (E&T) assistance to Member States (MS) in nuclear safety. The strategy is evolving and being successfully implemented. A methodology was developed for identifying knowledge gaps based on competency frameworks. Extensive education and training multimedia material based on the IAEA safety standards has been produced and made available to the Member States. Documents, lectures and other nuclear installation safety E&T material can be obtained directly from the IAEA website (http://www-ns.iaea.org/training/ni/default.htm).

Assistance has been provided to perform training needs assessment, to identify gaps, to design safety-related training programmes and to implement training using the training material prepared by the IAEA and made available to MS.

A3.6 IAEA training resources on the Web

Within the IAEA, training materials and resources can be grouped as follows.

A3.6.1 Main training strategies in the thematic areas of the IAEA's work

The various IAEA departments dealing with nuclear safety and security, nuclear development, nuclear sciences and applications, safeguards, technical cooperation and offices such as the Office of Legal Affairs, have developed in some cases specific strategies for training support of the MS. These are usually collected on a main web page for public information under the department or office. In addition, there are dedicated pages for information systems or target groups. A specific site for safety infrastructure deserves special mention that was built to make available all tutorials and workshops related to embarking countries (see http://www-ns.iaea.org/tech-areas/safety-infrastructure/default.htm).

A3.6.2 Training materials

A variety of courses and materials are available in different formats on the Web (http://www-ns.iaea.org/training/ni/materials.sp). There are basic professional courses and masters' degrees at a postgraduate level that give an overview of nuclear safety and the IAEA safety standards. These are often in e-textbook format. Due to its importance, regulatory control is the subject of a specific textbook and materials (http://www-ns.iaea.org/training/ni/e-textbooks.asp).

A3.6.3 Exchange of experience, training workshops

The IAEA hosts technical meetings to discuss safety issues and share experience of the Member States in safety as well as in the implementation of the safety standards (http://www-ns.iaea.org/standards/). Training seminars are also held in other organizations in cooperation with the IAEA. A number of these events are filmed and video presentations made available for 'training and exchange of information on safety issues. Some of this multimedia material can be found on the IAEA website (http://www-ns.iaea.org/training/ni/multimedia.asp).

A3.6.4 Training tools and services

A number of areas of work within the IAEA have developed specific tools and services. For instance, in the nuclear regulatory area, there is a knowledge gap analysis tool called SARCoN (Systematic Assessment of Regulatory Competence Needs) (http://www-ns.iaea.org/training/ni/tools-networking.asp) which includes a software application. This tool offers a competency framework for regulators to identify in a systematic way the competency gaps in their organization and help plan future training programmes or competence outsourcing. An example of such a service is 'train the trainers' based on the IAEA materials. This assists the trainers in using the IAEA training materials to design training programmes tailored to their organizational needs. These tools are available upon request addressed to the IAEA training coordinator of the Division of Nuclear Installations Safety (for further information please contact NSNI Training) and can also be requested through Technical Cooperation programmes which have supported tens of seminars based on the application of these tools.

A3.6.5 Training events and fellowships

A variety of training events and courses are held all over the world which the IAEA sponsors or organizes. Information on these events is available on the IAEA website (http://www.iaea.org/Publications/Training/index.html; http://ola.iaea.org/OLA/what_we_do/fellowship.asp). Proceedings and materials from the presentations are sometimes made available for information or self-study. Moreover, practical training is supported through fellowships in either the IAEA headquarters or other organizations.

A3.6.6 Support to regional cooperation and networks

Regional cooperation and networking is an efficient way of disseminating knowledge and facilitating training and accessibility to training materials. The IAEA cooperates in various regional networks that support knowledge in the area of nuclear safety (see the Asian Nuclear Safety Network (ANSN) below).

A3.7 The IAEA interdepartmental group on training and Web-based training resources

In 2008 an interdepartmental group on education and training was established in the IAEA in order to seek in-house harmonization and improvement of the training-related services to the MS. A working subgroup was set up in order to explore all the training resources available within the IAEA across the different areas of work, offices and departments. This group has identified IAEA web links under the above categories and compiled them into a centralized web page for training resources and materials (http://www.iaea.org/Publications/Training/index.html) where the user can find training-related resources as well as other knowledge database and information resources.

A3.8 Regional cooperation, knowledge networks and harmonized approach to training management

The Education and Training Topical Group (ETTG) of the Asian Nuclear Safety Network (ANSN) is an example of regional cooperation (www.ansn.org). This group consists of Asian countries willing to cooperate in the area of competence, knowledge and training for nuclear installations safety. Between 2006 and 2010, the ETTG built a general competencies framework (GCF) with more than 100 competencies in the area of nuclear safety based on the IAEA documents. The GCF identified different levels of knowledge (basic, medium and expert) and target audiences (regulators, operators, technical support organizations, and the general public). The ETTG then populated each of the areas of the general competencies framework with all the training materials and courses available in the Asian countries and in the IAEA and shared these on their web-based platform. Moreover, the

ETTG countries conducted systematic training needs analysis by performing competencies gap analysis against the general competencies framework. By analysing which parts of the GCF were relevant in their national situation and future plans, they then identified their national training framework (NTF) which they used as a basis for planning, training and prioritizing external assistance.

Finally, the ETTG share among their members experience and knowledge as well as all the training materials from the IAEA courses held in the participating countries. They are a good example of an experts' network, and they all benefit from the regional resources such as regional training centres.

A3.9 Conclusions and recommendations for efficient and sustainable training systems to build competence

Based on the experience from (1) supporting training and competence of human resources in countries interested in developing nuclear power programmes, (2) the application and seminars of the principles of the systematic approach to training (SAT) and SARCoN guidelines in more than 15 countries, (3) recommendations of the IAEA safety standards, in particular no. GS-G-1.1, *Organization and Staffing of the Regulatory Body for Nuclear Facilities* (IAEA, 2002), and (4) the research through questionnaires and analysis conducted by the Steering Committee of Competence of Human Resources for Regulatory Bodies, the following recommendations can be singled out:

- Establishing *a policy for training* and building competence. It is important that a policy is written, understood and followed by the concerned parties. The policy can be at a national level or for specific organisations. Commitment of resources, knowledge transfer and sustainability should be an important part of that policy.
- Use and build *national resources as much as possible* for training and conduct an analysis and planning for a *best optimal balance of regional and international cooperation and support to fulfil national gaps*.
- *Knowledge gap analysis and Systematic Training Needs Assessment (STNA)*. It is a first and necessary step to conduct an analysis of what is available and what is needed before starting to plan training and competence development. For some specific organizations such as an operator or regulator, there are well-defined competence frameworks. In the case of governments, competence frameworks, though still considered necessary, are often less developed.

- *Competence management tools and planning*. Once knowledge and competence gaps are identified, the next step is planning to fulfil the gaps. For that, scenarios must be developed taking into account the expected workload, workforce and available resources. When a plan is developed it is necessary to have in place knowledgeable individuals (training managers), processes and resources for implementing and assessing the plan.
- *Sustainability, ownership and knowledge transfer* should be a part of agreements and activities. One good practice is to train the trainers to use in an optimal way the comprehensive information and materials that are already available publicly to design effectively the training tailored to their organizations. The IAEA successfully provides workshops for adapting the IAEA training materials to trainers' needs.

A3.10 Acknowledgements

This appendix contains results of the work carried out not only in the division of nuclear installation safety but also in various IAEA training coordination and working groups. In particular, mention must be made of the excellent work of the training officers of the Nuclear Safety Training Group and the interdepartmental Education and Training Support Group/Web Working Group. Important highlights in the background and introduction come from the work of L. Lederman, consultant to the IAEA. I would like to thank especially G. Caruso, head of the regulatory activities section, for his support to the training activities and P. Woodhouse, senior regulator and safety officer, for his advice and revision of the content.

A3.11 References

IAEA (2001), International Atomic Energy Agency, *Training the Staff of the Regulatory Body for Nuclear Facilities: A Competency Framework*, TECDOC-1254, IAEA, Vienna.

IAEA (2002), International Atomic Energy Agency, *Organization and Staffing of the Regulatory Body for Nuclear Facilities*, IAEA Safety Standards Series no. GS-G-1.1, IAEA, Vienna (http://www-pub.iaea.org/MTCD/publications/PDF/Pub1129_scr.pdf).

IAEA (2006a), European Atomic Energy Community, Food and Agriculture Organization of the United Nations, International Atomic Energy Agency, International Labour Organization, International Maritime Organization, OECD, Nuclear Energy Agency, Pan-American Health Organization, United Nations Environment Programme, World Health Organization, *Fundamental Safety Principles*, IAEA Safety Standards Series no. SF-1, IAEA, Vienna.

IAEA (2006b), International Atomic Energy Agency, *Application of the Management System for Facilities and Activities*, IAEA Safety Standards Series, Safety Guide no. GS-G-3.1, IAEA, Vienna (http://www-pub.iaea.org/MTCD/publications/PDF/Pub1253_web.pdf).

IAEA (2006c), International Atomic Energy Agency, *The Management System for Facilities and Activities*, IAEA Safety Standards Series no. GS-R-3, IAEA, Vienna (http://www-pub.iaea.org/MTCD/publications/PDF/Pub1252_web.pdf).

IAEA (2008), International Atomic Energy Agency, *Milestones in the Development of a National Infrastructure for Nuclear Power*, IAEA, Vienna.

IAEA (2010), International Atomic Energy Agency, *Governmental, Legal and Regulatory Framework for Safety*, IAEA Safety Standards Series no. GSR Part 1, IAEA, Vienna.

INSAG (2008), International Nuclear Safety Group, *Nuclear Safety Infrastructure for a National Nuclear Power Programme Supported by the IAEA Fundamental Safety Principles* (INSAG-22), IAEA, Vienna.

A3.12 List of abbreviations and acronyms

ANSN	Asian Nuclear Safety Network
ETTG	Education and Training Topical Group
E&T	Education and training
GCF	General competencies framework
HRD	Human resources development
IAEA	International Atomic Energy Agency
KSA	Knowledge, skills and attitudes
MS	Member States (of the IAEA)
NPP	Nuclear power programme
NTF	National training framework
SAT	Systematic approach to training
STNA	Systematic training needs assessment
SARCoN	Systematic Assessment of Regulatory Competence Needs

A3.13 Annex: Four quadrants competencies model based on TECDOC 1254 and SARCoN guidelines

A3.13.1 Competencies related to legal basis and regulatory processes

This list includes competencies associated with both the legal basis and the regulatory process under which the regulatory body operates (guidelines available under http://www-ns.iaea.org/training/ni/tools-networking.asp). Legal basis competencies include those related to nuclear and other relevant legislation, decrees and regulations of the central government and local jurisdictions. Regulatory process competencies comprise knowledge, skills and attitudes (KSA) related to regulatory policies, procedures and other regulatory guidance documents as well as licensing documents that the staff members employ to carry out their duties.

These competencies and the associated KSAs include the following:

- *Legal basis competency*. The ability to read, comprehend, interpret and use relevant documents that establish the legal requirements for obtaining a licence, and the powers of the regulatory staff and the limits to these powers.
- *Regulatory process competency*. The performance of work in accordance with rules, regulations and established regulatory protocol to achieve the relevant regulatory objectives.
- *Regulatory guidance documents competency*. The capacity to produce regulations and guidance documents including policies and procedures containing practical steps on how regulatory requirements could be satisfied by the licensees and be adjudicated by the regulatory staff.
- *License and licensing documents competency*. The capacity to ensure that the licence and the associated licensing documents comply in form and contents with the regulatory requirements. This competency is related to a concept used by some regulatory bodies known as the safety case or safety envelope, which is normally defined by a licence and the associated licensing documentation.
- *Enforcement process competency*. The provision of a supportable recommendation of enforcement action in accordance with regulatory body policy.

A3.13.2 Competencies related to technical disciplines

This section addresses competencies associated with technology in various fields and areas that are needed by the regulatory body to carry out its overall responsibilities.

- *Basic technology competency*. Comprehension of science and engineering fundamentals in a particular field equivalent to a university degree. Examples are:
  - Nuclear engineering
  - Nuclear physics
  - Chemical engineering
  - Material science
  - Mechanical engineering
  - Civil engineering
  - Earth sciences
  - Environmental engineering
  - Computer science
  - Electrical engineering.

- *Applied technology competency*. Additional comprehension and demonstrated ability to apply engineering and science concepts in relation to the nuclear industry. Some typical applied technology areas for which many regulatory bodies provide technical training for regulatory body staff are listed below. Regulatory bodies commonly provide such training to generalists to broaden their competencies in specific areas. Regulatory bodies sometimes also provide such training to specialists in areas other than their speciality to broaden their perspectives of how their speciality area relates to other areas for which the regulatory body has jurisdiction. Examples are:
  - Reactor technology
  - Fuel cycle technology
  - Engineering techniques or technical issues
  - Radiation protection as applied to nuclear facilities and to industrial uses of radioactive sources
  - Nuclear safety technology including safety and risk analysis.
- *Specialized technology competency*. Comprehension and demonstrated ability to address and resolve issues in a specialized field. Some typical scientific fields or specialized areas that are common to many regulatory bodies are listed below. It should be noted that this is a sample list only and that a particular regulatory body may require competencies in other science and engineering areas:
  - Instrumentation and control
  - Criticality analysis
  - Nuclear material control
  - Software reliability
  - Fire protection
  - Human performance engineering/human factors
  - Fracture mechanics
  - Corrosion chemistry
  - Thermal hydraulics
  - Health physics.

A3.13.3 Competencies related to regulatory practices

- *Safety-focused analytical techniques competency*. The objective analysis and integration of information using a safety focus to develop a supportable regulatory conclusion.
- *Inspection techniques competency*. The independent gathering of information through objective review, observation and open communications, and determining the acceptability of information by comparing it to established criteria.

- *Auditing techniques competency*. The review of documents and/or programmes for conformity to established standards and procedures and making recommendations based on the results.
- *Investigation techniques competency*. The pursuit of the cause of events arising from notifications, incidents or information obtained during inspections and/or evaluations and gathering evidence in order to make regulatory decisions.

A3.13.4 Competencies related to personal and interpersonal effectiveness

This section addresses competencies associated with the personnel and interpersonal effectiveness of regulatory body personal while carrying out regulatory activities either individually or as part of teams.

- *Analytical thinking, problem-solving and decision-making competency*. Approaching problems objectively, gathering and integrating information, and developing a comprehensive understanding to reach conclusions.
- P*ersonnel effectiveness competency*: information technology + planning and organization of work + self-management competencies. Using technology to create, gather, manipulate, communicate, and/or share information. Effective and efficient coordination of tasks to achieve a desired objective. Working independently, exercising judgement and exhibiting flexibility in the completion of activities, especially during difficult or challenging situations.
- *Communication competency*. Engaging in effective dialogue, representation and interaction with others through committed listening, speaking, writing or delivery of presentations. Understanding the true interests of people and delivering meaningful understandable messages.
- *Teamwork competency*. Working in collaboration with others to achieve common objectives.
- *Management competency*: leadership + negotiation + project management competencies. Exemplifying by practice tolerance, objectivity, openness and fairness in dealing with colleagues and subordinates; dealing with stakeholders to achieve a consensus view over a strategy or programme of actions to achieve safety improvements; completing a set of complex tasks in a coordinated manner to preset time, scope and budget.

Appendix 4

Simulator training for nuclear power plant control room personnel

E. LINDAUER, formerly Director of the German Simulator Centre (GfS), Germany

Abstract: Training at a full-scope simulator is an indispensable part of the training of the shift crews of nuclear power plants. This appendix discusses the competences these personnel need, the content of the training and the elements which make up good training. Important elements are systematic planning, well-trained instructors, a focus on shift cooperation, assessments of the trainees and the programme, a suitable simulator and strong operations management involvement. Modern training simulators can be used for many purposes in addition to training.

Key words: simulator training, control room personnel of nuclear power plants, full-scope simulator.

A4.1 Reasons for simulator training

The direct operation of a nuclear power plant (NPP) is done by the main control room crew. Their actions are in accordance with operation plans, management directives and plant procedures, but the immediate action is taken by the shift personnel. Actions could be observation and interpretation of variables, switching systems and components on and off, taking controllers in automatic or manual mode, changing set points of controllers, adjusting valve positions, releasing work to be done on systems and accepting them back for operation afterwards. A minor part of the work is not done in the control room but directly at the systems, but this, too, is under the direct control of the control room personnel.

The type of work to be done is very different: the plant may be in normal operation, there could be a necessity to react to unexpected failures or malfunctions of systems or components, and the crew must also be able to cope with incidents and accidents.

- Evolutions in normal operation are planned and prepared for, i.e. as needed, the work and the procedures to be used can be talked through beforehand, and potential difficulties and risks discussed so that everybody has a clear picture of the imminent operation. Tasks in normal operation vary from simple one-actuation/one-effect actions to long sequences of interdependent operations of many systems, like the start-up of a plant from the cold shut-down state or the shut-down from power operation.

- Malfunctions of components and systems occur unexpectedly. Nevertheless there is a preplanned reaction to them. There may be some automatic response and in any case there are instructions for the actions of the crew. They have to make sure what the problem is and then determine whether the failure has an effect on the overall operation of the plant, whether such an effect can be averted, whether a failed function can be repaired or replaced, whether operation can be continued with or without changes in operation or restrictions or whether the plant has to be shut down.
- There can be serious incidents and accidents. The immediate response to these is automatic. The reactor will be shut down and safety systems started as needed to bring the plant into a safe state for a sufficiently long time. There will be lots of changes in plant parameters and numerous alarms going off. The crew has first to verify that a limited number of critical functions are performed, like the reactor is shut down and fuel cooling is available, which can be verified by checking about 25 parameters. This includes the control that the required automatic actions have taken place and the initiation of manual actions if needed. They then have to diagnose in which type of state the plant is (e.g. leakage or not from the primary coolant system, how is the residual heat removed, what is the actual heat sink) in order to determine the further operation to transfer the plant in a permanently safe state. This may but needs not necessarily include the identification of the fault that caused the event. Based on this information the appropriate operating strategy is determined and put in practice. During further operation they have permanently to monitor that the strategy chosen was the correct one and serves its purpose and that no new difficulties occur which have to be taken into account. All these diagnostic and operative actions are supported by emergency operating procedures and different information systems depending on the layout of the control room.

To perform all these activities safely and reliably the shift needs a solid basis. Especially, the individuals need to be well trained and they need to work in an effective organisation. The plants are different as well as the control rooms and the shift organisations. So in detail the performance of the shift crews is different. There are, though, constant features found in every plant.

There is one person in charge of the management of the shift crew. This individual takes the necessary decisions and gives directions to the shift. This person, called, for example, the Shift Supervisor, is in most cases permanently in the control room and performs this function during all operating situations. But there are other forms of organisation. For example, there may be a person who works near the control room and takes over control only when needed, notably when difficulties arise. There may also be a person on quick call who does not take over control but acts as an advisor to the Shift Supervisor. Most shift organisations provide, especially for the case of incidents and accidents, for more than one person in the shift coordination function so that they can share the workload arising from directing the plant operation and communicating with parties outside the control room.

The personnel described in the last paragraph do not directly manipulate the plant. This is done by typically two or three operators. Normally each of them is assigned a group of systems – which corresponds to a certain section of panels in the traditional control room design. They monitor these systems and perform switching operations as needed under the direction of the Shift Supervisor. Depending on the design of the control room there may be besides the main panels a number of boards where, for example, systems may be presented in more detail than on the main panel, so that single components can be switched. There may be an operator roaming in front of these boards under the direction of either the Shift Supervisor or an operator.

The expectation of course is that the shift handles all the situations discussed above safely and professionally as if they did it every other day. An indispensable prerequisite for this is a broad and solid knowledge of the plant, its behaviour, the operating strategies laid down in the procedures and the documents at hand for the use of the shift. But knowledge is not enough. The crew members need to act professionally and with certainty. The difference between a person who knows how a challenging task has to be performed and a person who can do it reliably is experience. For many tasks of the operating personnel the experience cannot be gained through plant operation because the related situations are too rare. This is even true for some normal operation situations. For example, start-up from cold shutdown may happen in a certain plant only once a year. With six shifts some crew members may not have seen a certain phase for a decade. And many accidents that the crew is expected to handle safely have never occurred at any plant. Therefore the personnel need a facility where they can gain the necessary experience under conditions as real as possible, and that is the simulator.

A4.2 Deciding who should be trained in full-scope simulators

From the previous section it is clear that the control room personnel, the people directly operating the plant, are the most important group to receive full-scale simulator training. This includes the individuals who perform switching operations as well as the individuals giving directions to them. If the organisation stipulates that supervisors who normally work outside the control room are called in certain situations and then take control over the plant operation, these persons need to take part in the simulator training, too. The same holds if the person called in acts only as an advisor to the control room manager in charge.

For this group of personnel simulator training in a full-scale simulator is indispensable. They need to have been trained on the simulator when they assume their duties and to continue regular training as long as they perform them.

The control room crews commissioning a new plant have the same responsibilities as the crews of an operating plant. Well before nuclear operation they need to have the same competences. They therefore have to be trained at a training simulator. The training content can be different in certain areas, because commissioning offers experiences in the real plant which during normal operation can only be gained at a simulator.

It is highly advisable that plant managers in the direct reporting line of the control room personnel, e.g. Operations Manager, Plant Manager, etc., in addition to observing simulator training, also participate in simulator training. They do not need to be able to perform the task of the control room personnel in detail, but they should have a realistic picture of what the control room work looks like. This is necessary because the management sets the standards for the work in the control room, approves procedures, decides on improvements of the control room, training facilities, etc., i.e. they are responsible for the work environment of the control room crew. Many of these individuals have a control room working history of their own, but this may be some time ago. Management training of course would be specific and a frequency of once in several years would be appropriate.

Many utilities find a certain amount of simulator training useful for the plant engineers. It helps meet the needs of operation in engineering changes and facilitates communication between the two groups.

For authority personnel overlooking the operation of NPPs, some impression as to how a plant is operated is also advantageous and some simulator training is useful to this end.

Simulator training for NPP personnel has been used since the 1960s. In the early days there were training settings which would not meet today's expectations: there were simulators with only a general similarity to the plant, some models being overly simplified due to lack of computer power. Several means were utilised to compensate for the shortcomings. Since then the requirements have been significantly increased. At least in the last two decades, it is generally accepted that a systematic training at a full-scope simulator with a good representation of the plant is an indispensable element of plant orientation.

A4.3 Operating scenarios for training

In principle, all operating situations have to be trained which are postulated to be handled by the operations personnel, and where the proficiency of action is dependent on prior exercise which cannot be gained sufficiently during plant operation. This includes normal operation, component and system malfunctions, incidents and accidents, and beyond-design accidents. There will be some differences between initial and continuing training. In initial training the operation of all systems has to be learned, i.e. including routine actions which later on are sufficiently exercised at the plant itself. Also in order to understand the strategies for coping with incidents and accidents these situations will be first trained without any additional complications. But at the end of initial training the trainees have to meet the same expectations as their colleagues in operation. Especially, they need to be able to diagnose situations that are initially unclear; this is best trained by inserting additional failures, faulty instrument readings and other disturbing problems.

To assess whether performance is dependent on prior exercise, different approaches may be used. As an example the following criteria are mentioned which are used in the German simulator training centre. There, a need for exercise is stated for all situations which:

- the personnel may perceive as stressful (this includes all accidents within and beyond the design basis);
- are complicated, e.g. with regard to their physical background, the I&C involved;
- are sensitive, i.e. mistakes may have large consequences, or
- are rare in real operation.

For beyond-design accidents there are limitations for the usefulness of simulator training. A number of decisions will be taken by the emergency management on the basis of various analyses. Only the actions in the control room would be trained on the simulator if reasonable, e.g. venting of the containment by opening two valves after many hours is no candidate for simulator training. Also evolutions should be presented on a simulator only if their realism is assured.

A4.4 Competencies to be acquired

The basic competence the control room staff needs is a thorough technical understanding of the plant and its behaviour. In first learning this understanding, simulator training plays a supporting role. The basis has to be laid by classroom training, study of reports and other instructional means for knowledge transfer. But simulator training is then a good means to strengthen and deepen the knowledge by application and exercise. And certainly simulator training is important for maintaining the technical competence. There is no better way to realise whether one has really understood a matter than to solve real problems.

The members of a shift group must be able to work together efficiently as a team. This starts with simple things: they have to be familiar with the control room and its instrumentation and the documentation for operation. They have to follow routine expectations regarding work performance: use of procedures, checking that an intended switching operation is performed on the right device, application of double checking by a colleague if required, observing whether the intended effect results from an actuation. All of this has to be done in the same way in the real control room and at the simulator, but the simulator also offers opportunities to exercise it in more complicated situations.

Very important are management standards for the cooperation of the team. The communication is structured: the sender of a message has to address the receiver and deliver the message in an unambiguous way, normally using standardised wording; the receiver repeats what he has understood, which the sender then confirms. An order has to say clearly what should be done on what object, what type of feedback is expected and, if applicable, what should be paid attention to and what problems or risks may arise. The whole team has to know at all times what the present status of the plant is, which actions should be taken next and why. This is achieved by statements of the control room supervisor at intervals as required. All team members are required to announce any important changes in their working area. They are also required to tell when they are in doubt whether the shift supervisor's assessment is right. They should also point out when they see problems in the working area of a colleague which require an environment of giving and accepting critique.

Decisions on the further course of operation have to be taken by the shift supervisor, as far as possible by making use of the whole team's knowledge. Some plants have a formalised decision-making scheme which integrates the contributions of all team members. A special challenge is the phase immediately after the onset of a major disturbance of normal operation like an accident. There the status of the plant has first to be diagnosed in order to take reasonable action. In many plants there are schemes for what checks have to be done first and which information is to be collected in order to make this assessment.

Smooth and efficient cooperation of the team is an ambitious aim. While some of the elements described in the last paragraph can be pretty well formalised, others cannot. But they can all be taught and trained. Especially for this part of the performance, a thorough debrief after the training is important.

A4.5 Defining good simulator training

Good simulator training is a well-planned undertaking. Its contents are derived from systematic analyses of the tasks of the control room personnel. The same analyses are needed to generate the operating procedures for normal operation, malfunctions and accidents, instructions for alarm handling, limits and conditions for operation and other operating documents. During construction of a new plant this information has to be delivered by the plant supplier. Representatives of the future plant owner should be involved early on.

From the task analysis there results a list of training goals from which certain goals are taken that are to be achieved by simulator training. This leads to a list of operating scenarios that have to be part of the simulator training. It is advisable that for a new plant the identification of training goals and operating scenarios is done as much as possible by staff of the future plant owner in cooperation with the supplier. In an operating plant the maintenance and updating of all of these documents is the responsibility of the operations department.

The list of scenarios as described above denotes the content of the initial training. For the continuing training it has to be decided, in addition, which ones need to be refreshed and how often, e.g. the whole content of the continuing training can be repeated in a three-year cycle with some operating scenarios being taken only once in this time and others every year. The repeating periods are determined by experience, e.g. more complicated scenarios from which details may be forgotten easily have to be repeated more often. Repeating operating scenarios does not mean repeating the same exercises. It means that the training goals connected with this operating scenario are rehearsed with different exercises. The same approach which was used to develop the programme initially has to be maintained throughout its lifetime. Changes, e.g. in equipment, procedures or regulatory requirements, which may affect the training have to be analysed and if needed the training has to be changed accordingly.

In designing a specific simulator course more aspects have to be considered. Part of the exercises will come from the list mentioned above. Another part will cover operating experiences, especially incidents. Lessons from incidents in one's own plant have to be disseminated to all shifts soon after they occurred, especially if they indicate training gaps. Incidents in other plants will be preferably integrated into a course which covers related issues. It is a routine task for a simulator instructor preparing a course that he scans the operating experience for information which is useful for the training goals at hand. An easy access to the operating experience has to be provided. An important part of the training is the familiarisation of the personnel with changes in the plant and in procedures. There must be an organised way of getting this information for the simulator training. For plant changes this must be done with a sufficient lead time because the simulator itself may have to be adapted. A good course has certainly to pick up specific needs of the participating students. These may be performance or knowledge gaps detected during plant operation, rehearsal of weak points of the previous course or relevant requests by the trainees.

These contents are arranged into a course programme. Although of course at the simulator many more perturbations of the operation occur than in the real plant, the aim is still to give the trainee as much as possible the feeling that he is working in the power plant. There should be undisturbed phases, the operating situations should, if possible, be logically arranged one after the other, and lengthy phases should not be unnecessarily shortened by jumps; in short, evolutions which are impossible in the plant should be avoided if practicable. The evolution of the events has to be planned in detail. In particular, the expectations concerning what single trainees should do at what point of the scenario have to be defined and documented. Particularly on this point, there has to be a strong involvement of the plant management, since they must be sure that training received at the simulator is what is expected in the real control room. It is advisable for instructors and plant personnel to test a course at the simulator before presenting it to students in order to see whether everything works as expected, especially whether the expectations regarding the actions of the students are correct.

Good simulator training utilises a number of feedback loops. The most immediate is the briefing after a training session. The trainees discuss their performance, point out what they think went well, where they see areas for improvement and whether there are issues which need further clarification. The instructors and management observers add their observations, pick up issues which the trainees did not realise and give guidance as needed. Lessons learned from the debrief are very effective since they are directly connected to the recent personal experience. The effectiveness of debriefs can be supported very much by technical means and it is highly advisable to use them: quick access and display of training and technical material, replay of recorded parameters, videotapes of the control room, restart of the simulation at a defined point. The debrief should cover the shift performance with regard to technical and behavioural issues. The trainees should be given the opportunity to asses their performance. Instructors and management observers should take notes during the exercises to make sure that all relevant aspects are captured.

Another feedback loop is the evaluation of trainee performance for different reasons. In initial training the aim is to make sure that the students have achieved the training goals for the respective part of the training. It is also a goal to find out at an early time in the training process if an individual is not suitable for this particular job. The evaluation is done by observation during the training, oral or written tests and tests on the simulator. Consequences of negative evaluation results will in most cases be repetition of the items where gaps are detected, sometimes changes in the training programme and in rare cases taking individuals out of the training. In continuing training the main goal of the evaluation is to make sure that the personnel are prepared to perform their jobs to the expected standards. The best thing of course would be to evaluate this by observing their performance in plant operation, but this is only rarely possible because many capabilities can only be observed in situations which are very unlikely in reality. If the reason for a certain training content was the observation of a performance gap in the real plant, then it should be evaluated whether the gap was closed by the training. In most cases the evaluation can be done only at the simulator. The method can be by observing the trainees during the training sessions or performing separate examination scenarios. The evaluation of the trainees must be done in a systematic way according to a written procedure. The criteria must be well defined and based on observable facts. They must be known to the trainees and the evaluators need to be trained in their application. In a given course the criteria are linked to the expectations defined during the course preparation. The results have to be explained to the trainees and possibilities for improvement should be discussed if necessary. Trainees should also be given the possibility to challenge the assessment. Therefore the evaluator should document the concrete situations and actions on which the assessment is based in order to support an objective discussion. Actions resulting from the evaluation are normally a statement that the trainee is qualified for his job, often with recommendations in what areas he should try to improve further; in some cases remediation would be required and in rare cases an individual will not be allowed to continue with his job.

Another feedback loop is the evaluation of the effectiveness of the simulator training. Again the performance of the personnel in the plant is the most reliable indicator, but only part of the relevant performance is observable. A valuable feedback is the analysis of reports on incidents and other negative conditions in the plant. Their root causes have to be fixed and part of it is suboptimal training. What can be readily collected and evaluated are the direct feedback of the trainees and the critique of managers observing the training. A valuable source of information is provided by the assessments of trainee performance in the simulator training. They can tell something about the individual participant but also about the training. These assessments can be done at different times. So-called as-found tests, i.e. tests at the beginning of a training course, show the capabilities which the personnel have during their everyday work. These are, among others, the result of their simulator training programme. Tests at some time after the training can be targeted on the question of how durably specific training goals have been achieved. An integral view of the simulator training programme or parts of it can be done by systematic assessments, which also draw on the information sources mentioned above. These may be self-assessments or reviews by peers from other plants. For plants which are part of a training accreditation scheme such assessments occur periodically. Accreditation of training programmes is used by all US and some other utilities. Since it covers all training activities and not just simulator training, it is not elaborated here.

As mentioned before, the main purpose of simulator training is not to impart knowledge but rather to train its application in a realistic environment. But the simulator can be used very efficiently to support the imparting of knowledge. A split of time between simulator control room and classroom has proven to be useful. In the initial training the functioning of a system which was explained in the classroom can be immediately demonstrated in the simulator. In grasping why the simulator behaves as it does, one's own understanding is deepened and misunderstandings are prevented or corrected. In addition the link of a cognitive content with a visualisation strengthens the memory. In continuing training the same holds for new subjects, e.g. the functioning of a system after a major change. The larger part is maintaining the knowledge already acquired. Gaps can be recognised during handling a scenario. The individual will try to revive the missing knowledge. In the debrief these points will be taken up and clarified. While these are very useful activities, they are not sufficient to maintain systematic knowledge. Time has to be allocated to discuss in detail the knowledge areas identified to be needed for the job. Areas pertinent to the course programme should be selected. Whereas in the initial training the sequence would mostly be first the theoretical explanation and then the exercise, it should be the other way round in the continuing training.

The trainers have an essential role in good simulator training. There is normally more than one person with a similar or with different backgrounds. Very often there are individuals specialising in simulator training, mostly called simulator instructors, and individuals from the operating line management. Depending on the organisation, there may be differences as to who contributes which part of the capabilities discussed below. This is not differentiated in the following, but just the tasks are mentioned that the operations personnel have specifically.

Overall the trainers must be able to support the trainees efficiently in reaching the training goals. This implies that they are accepted by the trainees, i.e. the trainees are convinced that it is worthwhile and useful for them to listen to the trainers and follow their advice. A trivial requirement is that the trainers are able to operate the simulator professionally and make use of all its possibilities. Further, they have to be sufficiently familiar with the working situation of the trainees. If they have no operating background anyway, they have to spend some months in the control room. For the training as well as for the acceptance of the trainers it is important that they have a high professional competence. This implies technical capabilities, i.e. a thorough understanding of the plant and its behaviour and background and rationales for its design. It also implies behavioural aspects such as teamwork, leadership and communication.

In addition the trainers need to be proficient in methods to facilitate efficient training: to guide discussions, to put the right questions, to decide when to interfere and when not, to activate the initiative of the trainees, to observe and objectively evaluate the performance of the trainees, to criticise trainees and accept their criticism, and to facilitate an efficient debriefing. In addition to these capabilities needed during the training, the trainer has to be able to prepare the training. This implies selection and composition of scenarios, incorporation of operating experience, preparation of training material, and defining the expected trainee response.

A line manager of operations may contribute to different parts of the training. But his main task is to make clear what benefit the management expects from the training and what standards of performance they want to see followed in the control room. There must be no doubt that what is trained at the simulator is exactly what is expected in the control room. This message can be conveyed in the opening of the training, by challenging deviations during the training and by contributions during the debriefing. Of course, deviations from the expected standards have to be challenged in the same way during normal work.

For utilities which construct their first nuclear power plant the availability of good instructors may be a challenge. The plant owner has to build up a training organisation well before taking over the plant in order to take care of training during operation. As regards simulator training, the instructors need to meet requirements in three areas: subject matter knowledge about the plant, instructional capabilities, and familiarity with the simulator.

One good possibility would be to employ individuals with instructional experience in other plants. They would join the training courses the plant supplier has to organise for the future operating personnel. A good opportunity to intensify the plant knowledge is the cooperation in the simulator development project, especially in testing. This gives also familiarity with the simulator, which in any case is the least time consuming of the capabilities required. For support in plant knowledge it is advisable to arrange for experienced supplier personnel to stay at the plant after the start of commercial operation, which is also useful for purposes other than training.

If individuals without former instructional experience are to be trained to become instructors, cooperation with other utilities or utility organisations is advisable. Future instructors have to attend courses on instruction and have to practice as an assistant instructor in a sufficient number of simulator courses. Their training for plant knowledge is as for the instructors with previous experience.

The simulator itself is obviously an important element of good simulator training. The requirements it has to meet are considered separately under the next header.

A4.6 Requirements for simulators

The general requirement for a training simulator is that all training goals can be reached in using it. This implies that not every detail has to be identical to the real plant. But since tolerable deviations pertain only to peripheral systems, this is neglected in the following discussion. Similarly the accuracy of the parameters displayed is not an end in itself. It has to be appropriate to support all training goals. The simulation has to be accurate enough that the parameters are plausible to the operating personnel, that they are able to assess the plant situation correctly and that the correct plant responses result. This can often be achieved by defining a certain tolerable deviation, e.g. 0.5%. In certain cases this is not enough, e.g. the main coolant temperature in a PWR is controlled in normal operation in a narrow band. A deviation of centigrades would not be plausible to the personnel and is therefore not tolerable in the simulator even if it would not change the overall plant behaviour too much. Another example is that relatively small differences in parameters can result in different plant reactions, e.g. either one of two safety actuations with different consequences could be reached first. If it is always the same in the plant, it must be the same in the simulator.

In summary, the simulator has to look like the control room, is operated like the plant, and responds in all regards perceivable for the personnel like the plant in all operating situations which are needed for the training. As regards visual perception this is achieved by building a replica of the control room with its panels and instrumentation.

The plant behaviour is produced by integrating mainly software models of the single systems and possibly hardware equipment (e.g. controllers) into one coherent plant model. It is important to keep in mind that these are models. They should be based as much as possible on first principles, i.e. they should reflect the underlying physical processes. This gives the best prerequisite that they will work appropriately over the entire range of operating situations needed. But still they can take into account more or fewer details of the underlying processes and they rely on a number of numerical coefficients, like heat transfer coefficients, which are of course known only with a limited accuracy. So it cannot be taken for granted that a good model will show correct results in every case. This has two consequences. Before building a new simulator or replacing models in an existing one, the phenomena which are expected to be correctly reproduced need to be analysed and documented from a training point of view. The other consequence is that all important operating scenarios have to be tested extensively. In addition to model properties, simple mistakes, which are inevitable in a large software project, influence the result and necessitate thorough testing. In summary, in a simulator the laws of nature apply only as far as they are correctly implemented and therefore in a strict sense only those results are reliable which have been verified to be correct.

The simulator has to be kept at a high fidelity with respect to the plant. To this end modifications have to be performed. There are two reasons for modifications:

- Simulation deficiencies are detected throughout the life of a simulator.
- Changes are made on the plant.

Both have to be collected and implemented in the simulator. For plant changes there has to be a systematic screening process which analyses each change with respect to its training relevance and leads to the decision whether a plant change is to be represented in the simulator or not. In implementing the changes, high quality software standards have to be applied to make sure that no degradation of the simulator is caused by the changes, and especially that no deficiencies are introduced. This means the requirements of the modification have to be defined, the software developed according to the standards for high quality software development, the modification performed, its result tested and the changes documented. Since changes sometimes may have unexpected side-effects which are not detected through direct testing of the single modification, integral tests of the simulator have to be performed about once a year. To this end a set of standard scenarios is run which cover a good deal of the operating situations needed for training. The results are compared with previous validated tests. If the modifications have to be distributed over a longer time, priorities should be defined according to training needs.

The simulator is a training tool. It therefore has a number of features which allow the instructor to operate it in a way that best supports the training goals. There must be a number of initial conditions to start the simulator from, which should be easily adaptable to simulator modifications in order not to become obsolete. It must be possible to introduce different types of malfunctions, to freeze the simulation, jump back in time, replay some part of the simulation, accelerate slow processes, and record data for later discussion. With the current computer power, simulator manufacturers are able to offer instructor stations with a large range of functionalities. The instructors should be provided with flexible means to generate good training. This enhances training efficiency by saving instructor and trainee time and it adds to the motivation of the trainees who appreciate having a quality tool for their training.

There are a number of properties of a simulator which in the first line do not affect the training but the technical quality of the simulator. These are properties like reliability, maintainability and documentation of the model design. Economically it is worth considering carefully whether a higher investment pays because of lower operating costs and higher longevity of the simulator. In the second line these properties may also affect training if they cause downtimes of the simulators, cause difficulties in performing changes or fix deficiencies timely and correctly. This also influences negatively the acceptance of the simulator by the trainees.

The personnel that commissions a plant and starts its power operation needs the same qualification as the personnel operating a plant which is already years in operation. Among others they have to train on a simulator. If the plant under construction is sufficiently similar to an already operating plant, the simulator of the operating plant can be used. If this is not the case the simulator for the new plant must be ready for training in time, which means about one and a half to two years before nuclear operation. This is not easy to achieve. The design of the simulator has to start at about the time of the start of construction of the plant or before. Throughout the design of the simulator, many design details and other data of the plant needed for the simulation will not be available or may change. The simulator may even be used to optimise the I&C design, possibly delaying the time the simulator will be finished. All this means that the simulator design is an iterative process which needs a very tight connection with the plant project. If feasible, the cooperation of the future operating personnel with the simulator development offers valuable learning opportunities. Especially, the testing of adaptations of the simulation to data from commissioning and to changes performed due to experiences gained from commissioning adds to a thorough understanding of the plant. This is no replacement for an initial simulator course similar to that of the personnel of an operating plant, which has to be done in addition. Even at that time the instructors preparing and delivering the course have to keep in mind what uncertainties the simulation still has. When the plant is in operation the simulator will need an adaptation to the real data available only then.

A4.7 Other applications for training simulators

With the growing computer power the possibilities of utilising training simulators have been broadened and this will certainly continue. So the following examples are by no means exhaustive.

Simulators can support training in different ways. The use of stored data in the classroom to analyse the scenarios previously trained in the simulator was mentioned already. But instead of stored data a simulation can also be provided to support the classroom training. It can be used for analysis of scenarios trained before, for demonstration to complement classroom teaching or for other teaching purposes. The input and output can be done on control room panels displayed on screens or in many other ways, e.g. using graphs or animated pictures.

At the simulator, procedures are used extensively, and therefore existing procedures and especially newly developed ones can be tested there efficiently. This applies to the usability and practicability of different concepts to present the procedures but also to their correctness. So there is a wide range from just checking already developed procedures before their release for operation to utilising the simulator for developing them. If the procedures are computer based, there is also a variety of possibilities to link them with supporting information in a user-friendly way.

Similarly, man–machine interfaces can be tested and optimised at the simulator. If the interface is via software panels, different concepts of presentation can be flexibly analysed, but even hardware panels can be handled by means of virtual reality. In new power plant projects simulators are used to develop the man–machine interface.

Many applications of simulators are possible because of the sophistication of the models which allow simulating complicated phenomena with good accuracy. So if beyond-design evolutions of the plant can be simulated, emergency exercises with extreme scenarios can be performed with simulator support. Simulators can also be used for the analysis of events which occurred and also for safety analyses, e.g. for a safety case. In these applications it has to be carefully validated that the simulation is appropriate for the problem under consideration.

Simulators are also applied to support the design of plant systems, e.g. the functionality of I&C systems can be tested flexibly at a simulator. This can be done with a simulated version of the I&C or by using identical elements of the plant I&C, e.g. the original software for a digital system. In the latter case even part of the commissioning can be done at the simulator, thereby saving costly downtime of the plant. The simulator even has advantages over the plant since conditions can be simulated which are not feasible in the plant. Simulators are used to analyse design variants for other than I&C systems with a case-to-case validation as to whether the simulator response is adequate for that purpose.

A4.8 Conclusion

A full-scale simulator is an integral part of a nuclear power plant. It is necessary to operate the plant with acceptable safety standards. To make the best use of a simulator, it must be able to support all training goals, and the simulator training has to be thoroughly developed and efficiently delivered. Modern simulators offer benefits in applications additional to training.

A4.9 Sources of further information and advice

Use of control room simulators for training of nuclear power plant personnel, IAEA TECDOC-1411, September 2004
Contains examples from many countries of practices for organising and conducting simulator training, including procedures, training scenarios, instructor competence, debriefing and programme evaluation.

Authorisation of nuclear power plant control room personnel: Methods and practices with emphasis on the use of simulators, IAEA TECDOC-1502, July 2006
Contains development and examples of simulator examinations.

IAEA Technical Meeting on workforce planning for new nuclear power programmes, Vienna, 31 March to 2 April 2009
Contains contributions from various organisations on workforce planning, build-up and training.

Nuclear power plant simulators for use in operator training and examination, ANSI/ANS-3.5–2009, American Nuclear Society
Contains requirements on simulator capabilities, scope of simulation, testing and configuration management.

Appendix 5

Multinational Design Evaluation Programme (MDEP): multilateral cooperation in nuclear regulation and new reactor design

J. REIG, OECD/Nuclear Energy Agency, France

Abstract: The Multinational Design Evaluation Programme (MDEP) is a multinational initiative to develop innovative approaches to leverage the resources and knowledge of mature, experienced national regulatory authorities who are undertaking the review of new reactor designs. MDEP has evolved to a multinational cooperation programme that includes inspection activities and generic issues. Working groups are implementing the activities in accordance with programme plans with specific activities and goals, and have established the necessary interfaces both within and outside the MDEP members. Significant progress has been made over the past years on the overall MDEP goals of increased cooperation and enhanced convergence of requirements and practices.

Key words: Multifunctional Design Evaluation Programme (MDEP), multilateral cooperation, nuclear regulation, nuclear reactor design.

A5.1 Introduction

The Multinational Design Evaluation Programme (MDEP) is a multinational initiative to develop innovative approaches to leverage the resources and knowledge of mature, experienced national regulatory authorities who are undertaking the review of new reactor designs. MDEP has evolved to a multinational cooperation programme that includes inspection activities and generic issues. MDEP incorporates a broad range of activities including:

- Enhancing multilateral cooperation within existing regulatory frameworks
- Increasing multinational convergence of codes, standards, and safety goals
- Implementing MDEP products and regulatory practices to facilitate licensing reviews of new reactors.

A key concept throughout the programme is that MDEP will better inform the decisions of regulatory authorities through multinational cooperation, while retaining the sovereign authority of each regulator to make licensing and regulatory decisions.

The programme was initiated in 2005, and a planning meeting of the original 10 participating countries was held in June 2006. Initial efforts consisted of multilateral cooperation on the European Pressurized Water Reactor (EPR) design reviews, and a pilot project to assess the feasibility of enhancing multinational cooperation and convergence of codes, standards, and safety goals within existing regulatory frameworks. The multilateral cooperation on the EPR expanded on bilateral interactions that had already been established between France and Finland. A structure for the programme was developed consisting of a Policy Group to oversee the programme, and a Steering Technical Committee with Working Groups to implement the programme with the Nuclear Energy Agency (NEA) serving as the Technical Secretariat. In addition the International Atomic Energy Agency (IAEA) takes part in the work of MDEP. The last version of the Terms of Reference for the programme has been approved in January 2011 (MDEP, 2011).

MDEP includes two lines of activities developed by working groups, the Issue Specific Working Groups (ISWG) and the Design Specific Working Groups (DSWG), under different terms of reference (MDEP, 2010a; MDEP, 2010b).

The original programme of work was agreed at the Pilot Project and consisted of 10 activities and support procedures which were chosen because they could be accomplished in the near term, and would result in significant benefits while requiring minimum resources (MDEP, 2008).

A5.1.1 Activities

1. Undertake a multinational vendor inspection programme.
2. Complete the evaluation of the similarities and differences among codes and standards for pressure boundary components.
3. Evaluate the similarities and differences in other codes and standards, beginning with a comparison of the digital instrumental and control (I&C) standards.
4. Complete the evaluation of the similarities and differences in the overall scope of the regulatory review and analysis for severe accidents.
5. Compare how top-level safety goals are derived and expressed, and how achievement is judged among the participating countries, and determine the extent to which they can be considered equivalent.
6. Compare the approaches used for taking account of operating experience in regulatory reviews for new reactors.
7. Develop a programme to collect, share, and use construction experience feedback in regulatory reviews.

A5.1.2 Support procedures

8. Develop a legal framework and the necessary agreements that will support the free exchange of information, including the results of independent analysis and research, among MDEP participants.
9. Establish working groups to maximize interaction and cooperation among regulators during the planning and conduct of new reactor design evaluations and construction oversight.
10. Establish a 'library' to collect and share regulatory documents of common interest related to design review and inspection of new reactors. Support the document collection by developing a model for a description that can be included in, or added to, regulatory documents so that it is possible to understand the regulatory review performed and the decision reached.

A5.1.3 Implementation

Working groups are implementing the activities in accordance with programme plans with specific activities and goals, and have established the necessary interfaces both within and outside the MDEP members. Significant progress has been made over the past years on the overall MDEP goals of increased cooperation and enhanced convergence of requirements and practices. Accomplishments to date provide confidence that the MDEP structure and process is an effective method of accomplishing increased cooperation in regulatory design reviews. The progress that has already been achieved demonstrates that a broader level of cooperation and convergence is both possible and desirable. In March 2009, the MDEP Policy Group agreed that the programme must continue beyond the original two-year mandate to fully achieve the established goals. Therefore, MDEP is considered a long-term programme with interim results. Interim results are those products that document agreement by the MDEP member countries and are necessary steps in working towards increased cooperation and convergence.

A5.2 Programme goals and outcomes

The main objectives of the MDEP effort are to enable increased cooperation and establish mutually agreed upon practices to enhance the safety of new reactor designs. The enhanced cooperation among regulators will improve the effectiveness and efficiency of the regulatory design reviews, which are part of each country's licensing process. The programme focuses on cooperation and convergence of regulatory practices that will lead to convergence of regulatory requirements. Cooperation will allow a better understanding of each other's processes to encourage and facilitate eventual convergence. The goal of MDEP is not to independently develop new regulatory standards, but to build upon the similarities already existing and on existing harmonization in the form of IAEA and other safety standards. In addition, the common positions developed in MDEP will be shared with IAEA for consideration in the IAEA standards development programme.

MDEP is meeting its goal of enabling increased cooperation through the activities of the working groups. MDEP has been very successful in providing a forum for regulatory bodies to cooperate on design evaluations and inspections. In addition to organizing working groups, MDEP has provided each regulator with peer contacts who share information, discuss issues informally, and disseminate information rapidly. For example, the design-specific working group members have benefited significantly from the sharing of questions among the regulators, resulting in more informed, and harmonized, regulatory decisions. MDEP members have also been highly successful in coordinating vendor inspections in which the regulators share observations and insights. MDEP has made improvements in communicating information regarding the members' regulatory practices through development of an MDEP library which serves as a central repository for all documents associated with the programme.

MDEP is meeting its goal of convergence of regulatory practices by establishing common positions in both the issue-specific and design-specific working groups. The working groups are making comparisons of the regulatory practices in the member countries, identifying differences, and developing common positions. The working groups are also working with codes and standards organizations to identify differences and propose areas of convergence. MDEP has identified similarities and differences in inspection practices, and plans to develop a common MDEP vendor inspection procedure to be used for multinational vendor inspections.

Progress towards harmonized regulatory practices and requirements for Generation IV reactor designs will be a natural outgrowth of this programme, as the participating regulatory authorities find that multinational cooperation and convergence of regulatory practices become routine elements of their planning and execution of new design evaluations.

MDEP has been successful in meeting the expected outcomes as defined in the MDEP Terms of Reference (MDEP, 2011, expected outcomes) by increasing knowledge transfer; identifying similarities and differences in the regulatory practices; increasing stakeholders' understanding of regulatory practices; and enhancing the ability of regulatory bodies to cooperate in reactor design evaluations, vendor inspections, and construction oversight, leading to more efficient and more safety-focused regulatory decisions.

A5.3 Programme implementation

A5.3.1 Membership

Participation in the Policy Group (PG) and Steering Technical Committee (STC) is intended for mature, experienced national safety authorities of interested countries that already have commitments for new build or firm plans to have commitments in the near future for new reactor designs. Current MDEP members are the regulatory authorities of Canada, China, Finland, France, Japan, Korea, Russian Federation, South Africa, the United Kingdom and the United States. The OECD Nuclear Energy Agency provides the technical support to all MDEP activities. In addition the IAEA takes part in the work of MDEP.

A5.3.2 Organizational structure

The programme is governed by a PG, made up of the heads of the participating organizations, and implemented by a STC and its working groups. The STC consists of senior staff representatives from each of the participating national safety authorities, plus a representative from the IAEA. The NEA performs the Technical Secretariat function in support of MDEP.

The PG provides guidance to the STC on the overall approach, monitors the progress of the programme, and determines participation in the programme.

The Steering Technical Committee manages and approves the detailed programme of work, including defining topics and working methods, establishing technical working groups, and approving procedures and technical papers developed by the working groups; establishing interfaces with other international efforts to benefit from available work; developing procedures for the handling of information; reporting to the Policy Group; and identifying new topics for the programme to address.

Two lines of activities have been established to carry out the work:

- *Design-specific activities*. Working groups for each new reactor design share information on a timely basis and cooperate on specific reactor design evaluations and construction oversight. Participants in these working groups are the countries that are actively reviewing, preparing to review, or constructing the specific reactor design. A design-specific working group is formed when three or more MDEP member countries express interest in working together. Under the design-specific working groups, subgroups have been formed to address specific technical issues.
- *Issue-specific activities*. Working groups are organized for the technical and regulatory process areas within the programme of work. These currently include, but are not limited to, vendor inspections, pressure

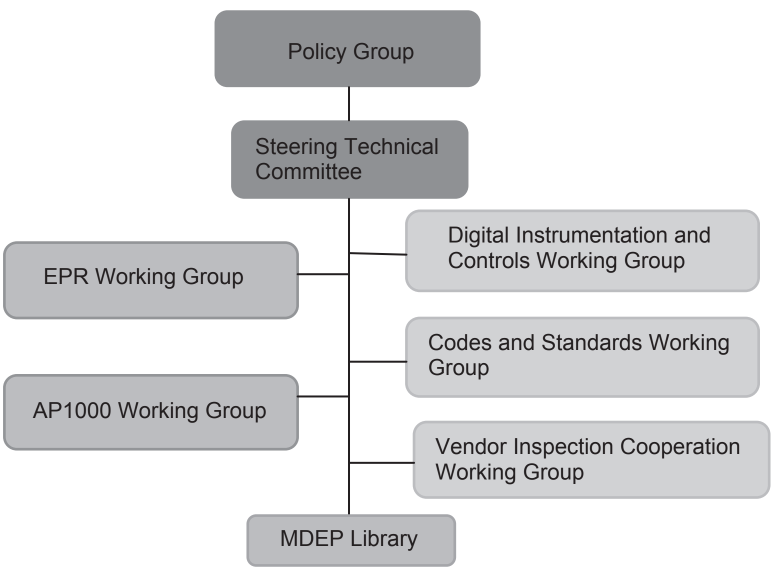
boundary component codes and standards, and digital instrumentation and control standards. Membership in issue-specific working groups is open to all MDEP participating countries and the IAEA representatives.

Figure A5.1 illustrates how the programme is organized.

A5.3.3 Communications

MDEP information is communicated among the members through the MDEP library which serves as a central repository for all documents associated with the programme. NEA provides the technical support for development and maintenance of the MDEP library on a website. The website includes a folder structure and provides for two levels of access which are password protected: (1) MDEP member countries, and (2) member countries participating in design-specific working groups. Access to the library is based on requests of the STC member for each participating country and generally consists of the STC members and members of the working groups. Publicly available documents related to MDEP are available on the MDEP page of the NEA website (www.oecd-nea.org/mdep/).

Each MDEP member country has designated a contact within their regulatory organization with the responsibility to identify documents to be included into the library, track library status and associated activities, and maintain contact with the NEA librarian. NEA has issued a guidance



A5.1 Organization chart of the MDEP.

document detailing library functions, access, and use. The library documents are either in English or include an abstract in English describing the contents. NEA is pursuing a process for translating documents.

In order for MDEP to be successful at fulfilling its goal of leveraging the work of peer regulators in the licensing of new nuclear reactor designs, a framework was developed to facilitate the sharing of technical information among MDEP participants which at times may include the sharing of proprietary and other types of sensitive information. As a general rule, the information exchanged as part of the MDEP in meetings and the MDEP library is for use only by the participating national regulators. The members of the design-specific working groups also have a communication protocol to share MDEP positions on topics with other members in advance of release of this information into the public domain. A large portion of the information shared may not be proprietary or sensitive and therefore easily accessible, mainly those generated within the issue-specific working groups; however, all participating members must protect and properly handle the information that an originator claims to be proprietary or sensitive.

A5.3.4 Interaction with industry groups

The MDEP working groups are very interested in understanding the perspectives of the design vendors, codes and standards organizations, and component manufacturers in the MDEP activities, and the challenges they face in dealing with numerous regulators and regulatory systems. The MDEP working groups interact with, and invite industry groups to participate in selective portions of meetings and other activities. For example, the EPR Working Group interacts with AREVA; the Codes and Standards Working Group is interacting with a committee of standards development organizations (SDOs) (American Society of Mechanical Engineers (ASME), Japan Society of Mechanical Engineering (JSME), Korea Electric Power Industry Code (KEPIC), Règles de Conception et de Construction des Matériels (RCCM), Canadian Standards Association (CSA)) in a code comparison project; the Vendor Inspection Cooperation Working Group invited several vendor companies to its meetings to make presentations of the vendors' perspectives of the regulatory requirements regarding pressure-containing components; and the Digital Instrumentation and Controls Working Group issued letters to International Electrotechnical Commission (IEC) and, Institute of Electrical and Electronics Engineers (IEEE) encouraging their continued cooperation on MDEP initiatives. In addition, the MDEP Policy Group is looking for opportunities for effective cooperation with the World Nuclear Association (WNA), Cooperation on Reactor Design Evaluation and Licensing (CORDEL) Working Group, mainly interested in defending the benefits of international standardization of nuclear safety standards for reactor designs (WNA, 2010).

A5.3.5 MDEP conference

In addition to the interactions discussed above, MDEP held its first conference on 10–11 September 2009, in the OECD conference centre in Paris. The conference was organized by NEA to communicate the goals, achievements and plans of MDEP to regulatory authorities not participating in MDEP and to other interested parties including the nuclear industry, standards development organizations, and other multinational organizations including IAEA. The conference was attended by about 170 individuals from 23 countries and 11 international organizations. Presentations and round-table discussions are part of the MDEP open publications (MDEP, 2010c).

The conference included presentations and panel discussions on the activities of each of the MDEP working groups, industry initiatives for new reactor designs, and international organization initiatives. The participants in the panel discussions included MDEP participants, other regulators, standards organizations, and industry representatives. Broad support for MDEP efforts was expressed by all participants. Several participants presented their views or specific proposals for expanding MDEP efforts and for increasing communications with outside organizations. Some regulators of small nuclear programmes and several representatives of industry expressed their desire for some type of international or multinational approval of new reactor designs. Other regulators of small nuclear programmes and all MDEP participating regulators re-emphasized the importance of strong, independent national regulators who are supported in their decision making through enhanced cooperation with other regulators. In this context, MDEP provides a forum for enhanced cooperation through peer discussions and sharing of documents on research results, analysis and regulatory practices.

Some of the conclusions that came out of the conference were that MDEP is an effective and efficient method of pooling experts from different countries. It improves the design reviews and enhances the safety level, and its efforts should be continued. In addition, stakeholders have great expectations of MDEP, so MDEP should pursue improved information dissemination to external stakeholders.

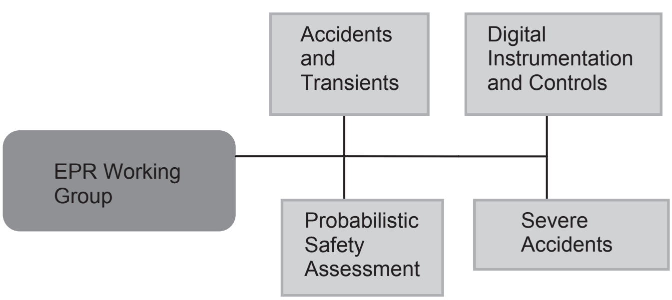
A5.4 Current activities

The current activities of MDEP are being implemented through design-specific working groups, issue-specific working groups, and subcommittees of the STC (MDEP, 2010d). The members of the design-specific working groups share information and cooperate on specific reactor design evaluations and construction oversight. Issue-specific working groups are organized for the technical and regulatory process areas within the programme of work. Each working group has a lead and co-lead country designated, and has developed a programme plan which identifies specific activities, schedules and contacts.

A5.4.1 EPR design-specific working group

The EPR working group currently consists of the regulatory authorities of France, Finland, the USA, UK, China and Canada. This working group was established in January 2006 as a multilateral cooperation between France, Finland and the US. Numerous meetings and technical exchanges have taken place to exchange information on the reviews being conducted in each country: Olkiluoto 3 (OL3) which is under construction in Finland; Flamanville 3 which is under construction in France; and the US version of the EPR which was submitted for design certification in the United States on 11 December 2007, and is referenced by four combined licence applications currently under review. In November 2008, China and the UK were added as members. China issued a construction permit for an EPR at the Taishan site in August 2009. The UK is performing a generic design assessment of the UK-EPR at the joint request of EDF and Areva. The design is essentially the same as the French design being constructed by EDF at Flamanville. Canada is in the first phase of its review of the EPR design application against the Canadian design requirements.

The goals of the working group are to reach convergence in aspects of the review of the EPR design where possible and find areas where member countries can cooperate. The activities of the working group are summarized in Fig. A5.2.



A5.2 Activities of the EPR Working Group.

Accomplishments

The EPR Working Group has been successful in identifying issues that were addressed by one country but not yet fully considered in other countries. For example, STUK (the Finnish regulatory authority) and ASN (the French regulatory authority) have shared portions of the detailed design of the EPR instrumentation and control system. This was useful to countries such as the US and the UK that had not seen the detailed design at that time. In addition, STUK shared the instrumentation and control issues identified in their review. The working group has also shared the resolution of issues by one country that may not have been fully considered in other countries. For example, the US shared its interim staff guidance for independence of data communications between various instrumentation and control systems.

The working group members have also discussed tools and methods used in their reviews that may be useful to other members. For example, the NII (the UK regulatory authority) discussed the use of statistical software testing as a demonstration for software meeting a particular reliability goal that can be used in the overall plant probabilistic safety assessment (PSA). Additionally, STUK provided a presentation of a software modelling tool that was used to evaluate the OL3 software and identified some requirements/design specification issues. This tool may be of value to other regulators when the software for the plants they are reviewing is under development.

The working group is maintaining in the MDEP library a listing of EPR technical issues that have been identified and are currently being evaluated by each of the participating regulators. The library provides a synopsis of the issues, the status within each technical body, and links to relevant documents.

Technical expert subgroups have been formed to address specific issues in the design review.

EPR Digital Instrumentation and Controls Subgroup

One notable accomplishment of this subgroup is the work the members completed in identifying a potential single-failure issue which led to design changes by AREVA. This is notable because the identification of the problem was made and shared pursuant to MDEP efforts and directly resulted in design changes. As a next step, the subgroup plans to coordinate audits of the I&C design process to evaluate the verification and validation process during the design phase.

EPR PSA Subgroup

Technical areas discussed by the subgroup include fire PSA, external hazards, and common-cause failures. Additional topics and documentation were identified for future information exchange. The working group members are continuing discussions of the differences between the French, Finnish and US designs and regulatory approaches. In particular, they are working to understand the differences in heating, ventilation and air-conditioning (HVAC) design and system interdependencies, and diesel generator battery design capacity.

EPR Severe Accident Subgroup

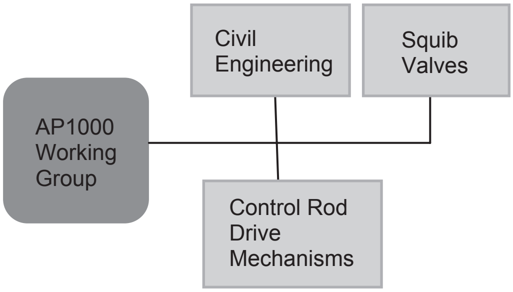
The Severe Accident Subgroup has discussed the use of the two-room concept in the containment response evaluation and molten core cooling system and structures. The subgroup compared the use of the codes that were utilized for various parameters and determined that there are some significant similarities, and some differences, among the approaches. The subgroup considers that the Operating Strategies for Severe Accident (OSSA) review is an important subject because of the new items specific to EPR (and not to currently operating PWRs).

EPR Accidents and Transients Subgroup

The first topics discussed in this subgroup have been containment response evaluations, accident analyses methodologies, and criticality safety during outages. The subgroup will continue to discuss containment response evaluations and the containment sump design issues (resolution of the generic safety issue and issues such as chemical effects on sump performance).

A5.4.2 AP1000 Design Specific Working Group

The AP1000 Design Specific Working Group was established in November 2008 with initial participation by the regulatory authorities of China (NNSA), the UK (NII), and the US (NRC). The Canadian regulatory authority (CNSC) was added as a member in March 2009. NNSA issued a construction permit in March 2009 for two AP1000 units at the Sanmen site. A total of four AP1000 units are planned for construction in China. The NRC is reviewing Revisions 16 and 17 to the AP1000 design certification and is concurrently reviewing combined licence applications for 12 AP1000 units. The Vogtle plant, for which NRC has issued an early site permit and Limited Work Authorization, is expected to be the first AP1000 to go into construction in the US. NII has completed Step 3 of the four-step generic design assessment process of the AP1000 design. CNSC has started the pre-project review on the potential choices for new reactor construction, including the AP1000. The activities of the working group are summarized in Fig. A5.3.



A5.3 Activities of the AP1000 Working Group.

Accomplishments

The working group has shared design information, application documents and preliminary findings within the group and identified the most significant review issues. Subgroups of experts have been formed to address specific technical issues that were identified by all participants as being significant because they involve unique or unresolved design features. Current subgroups include those for shield building, squib valves, and control rod drive mechanisms. Other topics to be discussed may include radiation protection, instrumentation and controls, and human factors. A status of the expert subgroups follows.

Shield Building Subgroup

The shield building design was selected for further discussion by an expert subgroup due to the uniqueness of the design, and the fact that there are currently outstanding questions regarding the modular construction techniques to be used and the use of former plates rather that rebar in the design of the concrete retaining walls.

The subgroup members compared results of their separate reviews of the shield building design and came to similar conclusions regarding fundamental concerns. The discussions were helpful in confirming conclusions already identified by the regulators. In the absence of applicable design standards for concrete composite structures, the expert subgroup developed a preliminary set of technical considerations to be used for novel civil engineering construction (such as modular steel composite structures). These considerations may be used to propose a code case to the standards organizations for modular construction.

Squib Valve Subgroup

The in-containment refuelling water storage tank injection valves (squib valves) were selected because of the uniqueness of these valves and their relative risk significance. Such valves are not currently in existence and will require a new design and associated qualification programmes. The squib valves to be used on the AP1000 are much larger than those used in existing nuclear applications. Questions have also been raised regarding the adequacy of the current in-service testing requirements for such valves, since there is little knowledge on operating experience. The members agreed that the lack of experience with large squib valves requires particular care in the design, qualification and in-service inspection/testing of these valves. The Squib Valve Subgroup has prepared an initial draft of technical guidelines for the design, qualification and in-service inspection/testing of explosive-actuated valves. The guidelines are intended to be helpful to regulators and the nuclear industry in understanding the technical issues associated with large explosive-actuated valves used in AP1000 reactors and other reactor designs.

Control Rod Drive System Subgroup

The control rod drive system was selected because its safety classification (classified as non-safety) has been questioned by China, particularly the classification of the latch mechanisms and the adequacy of any associated testing or analysis to show that the latch mechanisms can perform their intended safety function.

A5.4.3 Vendor Inspection Cooperation Issue Specific Working Group (VICWG)

Background

A Working Group on Component Manufacturing Oversight was established as part of the MDEP pilot project to assess the regulatory requirements and review associated with the manufacturing processes for components for use in nuclear power plants. The working group met with the design code bodies from the US, France, Japan and Korea, and found that component manufacturing is currently subject to multiple inspections and audits similar in scope and in safety objectives, but conducted by different organizations. The pilot project concluded that the formation of multinational regulatory teams to perform inspections of component manufacturers will improve effectiveness and efficiency in the regulatory assessment of the highest safety class components.

A working group was established to continue the work of the pilot project to identify areas of commonality and differences between regulatory practices of participating countries in the area of vendor inspection programmes. The long-term objectives of the working group are to maximize the use of the results obtained from other regulators' efforts in inspecting vendors, and to perform multinational inspections of vendors according to the common quality assurance requirements.

To improve the use of the result of other regulators' inspections, the working group will continue enhancing the understanding of each regulator's inspection procedures and practices by coordinating witnessed inspections of safety-related mechanical pressure retaining components (Class 1) such as pressure vessels, steam generators, piping, valves, pumps, and quality assurance (QA) inspections. The working group plans to develop and maintain a process to share inspection results, including a library of all inspection results. In the longer term, a process will be developed to adapt the scope of an inspection according to the need of other regulators.

In order to be able to perform multinational inspections, the working group will identify and document a set of common QA requirements (write essential elements for QA among MDEP countries). The participants may agree on an acceptable method to assess the implementation of the common QA requirements and then will develop an MDEP QA inspection procedure (documentation and training).

Accomplishments

The VICWG developed matrices that identify the scope of inspections in each country. Understanding which inspection areas are covered by each regulator helps the MDEP countries to coordinate vendor inspections, and will provide each regulator with a better understanding of the applicability of inspection findings by other countries.

The group is currently performing witnessed inspections, which consist of one regulator performing an inspection to its criteria, observed by representatives of other MDEP countries. Thirteen such inspections were conducted in 2009, in five countries and with the involvement of seven regulatory bodies. The VICWG maintains a 'vendor inspection planning table' with a list of scheduled vendor inspections to assist the member regulators in identifying opportunities to observe an inspection, or obtain the results of an inspection carried out by another member. The benefits to the observing countries include additional information and added confidence in the inspection results.

The working group developed an MDEP vendor inspection protocol document with guidelines for witnessed and joint inspections. This document will facilitate inspections that are observed and attended by multiple regulators.

The working group has initiated an activity that could lead to identifying common quality assurance requirements of the regulatory bodies. The group conducted a survey on quality assurance requirements used in the oversight of vendors to identify those areas where the various regulators have common regulatory frameworks. A comparison table has been drafted and will be finalized in 2010.

Next steps

In 2010, the working group plans to perform more witnessed vendor inspections. This will continue to enhance the exchange of information between the regulators and provide better understanding of the inspection scopes and safety findings and how these findings may be utilized. In order to improve the process for sharing inspection results, the working group will write a procedure to share inspection results, and improve the MDEP library to include an inspection results database.

In 2010, the group plans to develop and implement the common processes needed to adapt the scope of vendor inspections to take into account the needs of other member countries; and to develop a framework that will allow MDEP members to take into account other regulators' vendor inspections.

The next planned phase is 'joint' inspections which consist of one lead regulator and other MDEP members participating. This would allow the participating members to use the results of the inspection that are applicable to their regulations. To implement this, the WG will update the protocol for conducting joint inspections and identify training needs to support joint inspections. MDEP plans to organize at least two or three joint inspections by the end of 2011.

A5.4.4 Codes and Standards Working Group

Background

The primary goal of the Codes and Standards Working Group (CSWG) is to achieve convergence of regulatory requirements in the area of component design. A major initial step towards this goal is establishing a retrievable database of the similarities and differences among the codes and standards used in the design of pressure boundary components. The initial effort emphasized the similarities and differences among the codes and standards used in the US (ASME), France (RCCM), Japan (JSME) and Korea (KEPIC). Future efforts will address codes and standards in other countries including Canada (CSA) and the Russian Federation. The working group's goal is to perform an assessment of the similarities and differences for the codes and standards, and identify the most beneficial areas for convergence. Changes in codes and standards can only be made by the SDOs themselves and therefore the role of the working group is to assist the SDOs in identifying and resolving important differences. The goal of both the SDOs and the CSWG is to achieve global harmonization of pressure-boundary design codes for nuclear power plants.

Accomplishments

The CSWG has interacted with SDOs which have formed a steering committee composed of the representatives of ASME, JSME, KEPIC, AFCEN, CSA, vendors and utilities. The SDOs are performing a code-comparison project in conjunction with the working group's efforts. The first phase consists of a comparison of each code's requirements for Class 1 pressure vessels with those of the ASME code, Section III. This comparison includes the material, design, fabrication, examination, testing, over-pressure protection and general requirements. The SDOs have prepared a comparison table of the pressure boundary codes for Class 1 pressure vessels. This assessment was accomplished through correspondence and joint meetings between the working group and SDOs. The initial effort focusing on pressure vessel codes resulted in a database which identified the similarities and differences between the Korean, Japanese and French codes as they compare to the ASME code. The project was designed to use the ASME code as the basis for the comparison, since most of the codes under review originated from the ASME codes. The source of the differences in the codes, such as regulatory requirements or code organization approach, is also addressed. The Phase 1 code-comparison activity for the KEPIC, JSME and RCCM codes is complete. Canada and Russia have also initiated a code-comparison effort.

It has become clear that the complete convergence on every aspect of pressure-boundary codes on an international scale is not currently feasible because of the large differences in the scope of the different designs, each country's design and construction practices, regulatory requirements and processes, cultural patterns, and the manner in which codes are adopted by regulatory agencies. Based on the results of the comparison exercise, the CSWG has concluded that although full convergence of the codes is not feasible, harmonization is possible. The key to achieving harmonization is to understand the source of and reasons for differences of code requirements in order to assess their significance from a safety and risk perspective.

Based on comparison results of Class 1 pressure vessels, the working group has begun discussions to identify the sections of the codes that are equivalent or identical, and the sections that are not equivalent, and to examine potential paths for reconciliation of the differences in the codes, including identifying those that should be pursued for potential convergence. As an interim measure, the working group has obtained a commitment in principle from the SDOs to work together to minimize further divergence of code requirements.

Next steps

The SDOs are continuing their code-comparison effort for Class 1 piping, pumps and valves (Phase 2). This next phase is expected to be much simpler than Phase 1 because the general requirements and technical requirements for materials, fabrication, examination, testing and over-pressure protection, which are being completed in Phase 1 for Class 1 vessels, are also applicable to Class 1 piping, pumps and valves.

Once an understanding is gained of the differences between the codes, each MDEP participant could initiate their national process to endorse, in whole or in part, the pressure-boundary codes and standards of other countries. Also, the working group will continue discussions with the SDOs for finding potential paths for harmonization of the differences in the Class 1 vessel codes. Plans to further expand the scope of work to include Class 2 and 3 vessels, piping, pumps and valves will depend on the success of Phases 1 and 2. Ultimately, MDEP will expand the codes and standards harmonization effort to areas beyond pressure-boundary components.

A5.4.5 Digital Instrumentation and Controls Working Group

Background

The objective of the Digital Instrument and Controls Working Group (DICWG) is to identify opportunities for convergence of applicable standards. The working group's activities include identifying and prioritizing the member countries' challenges, practices and needs regarding standards and regulatory guidance regarding digital instrumentation and controls, identifying areas of importance and needs for convergence of existing standards and guidance or development of new standards, sharing of information, and developing the common positions among the member countries for areas of particular importance and need.

To enhance cooperation with the standards organizations, there is interaction with the IEEE and the IEC. Representatives from IEEE, IEC and IAEA participated in the working group meetings, and both IEC and IEEE allowed a number of their standards relevant to digital I&C to be made available in the MDEP library for use by the working group members. The IEC formalized an agreement with the OECD to facilitate cooperation between the two organizations. The working groups also interface with equipment designers and manufacturers to share their experience.

Accomplishments

The working group identified the member countries' most significant technical issues regarding standards and regulatory guidance related to digital instrumentation and controls. This list was used to better understand the main issues and to determine priorities for the working group and is reviewed on a periodic basis. The working group performed a comparison exercise to identify the similarities and differences in regulatory requirements applicable to these areas, and prioritized the differences that should be addressed for increased convergence work. In particular, the working group evaluated the key differences between the regulatory framework established in accordance with the IAEA guidance and IEC standards, and with the NRC requirements and IEEE standards. In all of the priority areas, the working group identified that there were significant similarities and overlaps in the regulatory approaches.

In addition, the working group compared the list of IEC standards and IEEE standards relevant to digital instrumentation and controls. A detailed comparison table has been developed and reviewed by the working group. This comparison resulted in significant findings regarding the standards in terms of the development status, scope and details as well as the differences and similarities at a high level. The working group engaged IEC and IEEE, as well as IAEA, regarding their participation in a comparison exercise of the standards and increased coordination related to digital instrumentation and controls. Based on the results of the comparison exercise, the working group issued letters to IEC and IEEE recommending that the standards organizations consider the MDEP common positions when revising their standards and increase their cooperation to achieve enhanced harmonization of relevant standards.

The DICWG developed common positions on specific issues among the member countries which are based on the existing standards, national regulatory guidance, best practices and group inputs using an agreed process and framework. To date, the working group has identified a number of areas for potential convergence and has been developing common positions. Two common positions on software common-cause failure and software tools are complete, and additional ones are under development with an expected completion date in early 2010. The common positions under development include independent verification and validation, data communication, simplicity in design, and complex electronics. Additional topics will be identified as the working group completes these common positions.

Next steps

The working group will continue to develop additional generic common positions as more common positions are identified and developed.

The working group will communicate specific suggestions to the standards organizations and IAEA for consideration of harmonization in a timely manner when they are identified during its activities.

The working group will continue to exchange information among members to contribute to efficiency and effectiveness of the licensing of new reactor digital instrumentation and controls.

The working group will continue to engage digital instrumentation and controls vendors and utilities to share experience and insights towards developing common positions that are based on a broad spectrum of inputs.

A5.4.6 Safety goals

Background

One of the original 10 recommendations of the MDEP pilot project was to compare how top-level safety goals are derived and expressed, and how achievement is judged among the participating countries, and to determine the extent to which they can be considered equivalent. MDEP has recognized that the route to harmonization of safety goals must start with high-level, mainly qualitative goals, which will not be dependent on the reactor technology considered. This understanding is expected to enhance cooperation in using other regulators' assessments and understanding of how decisions have been reached.

The objective of the task is to determine (1) how various countries describe the desired level of safety to protect public health and safety and the environment, (2) the role of deterministic and probabilistic considerations, and (3) other groups and organizations that are involved in similar or related work.

Accomplishments

One of the major outcomes of the work is increased understanding of the origin of the safety goals in several countries. The group is developing a framework paper, based on the defence-in-depth concept and probabilistic considerations. This framework can be useful for development of safety goals and support of safety decisions by safety authorities and designers.

Next steps

After the framework paper is finalized, MDEP plans to meet with other organizations and finalize its recommendations for high-level safety goals in a position paper. The work will be complete upon issuance of the position paper (scheduled for March 2011). The MDEP recommendations related to high-level safety goals will form the basis for MDEP contributions to the work being performed in this area by other national or international organizations.

A5.5 Interim results

In March 2009, the MDEP Policy Group agreed that the programme must continue beyond the original two-year mandate to fully achieve the established goals. Therefore, MDEP is considered a long-term programme with interim results. Interim results are those products that document agreement by the MDEP member countries and are necessary steps in working towards increased cooperation and convergence. The interim results so far include the following:

- Identification and documentation of technical expert subgroup technical reports that identify and document similarities and differences among designs, regulatory safety review approaches and resulting evaluations
- Agreement of a listing of EPR technical issues that are currently being evaluated by each of the participating regulators, including a synopsis of the issues, the status within each technical body and links to relevant documents
- Establishing a preliminary set of technical considerations to be used for novel civil engineering construction (such as modular steel composite structures) and technical guidelines for the design, qualification and in-service inspection/testing of explosive-actuated valves
- Maintaining a vendor inspection planning table with a list of scheduled vendor inspections to assist the member regulators in identifying opportunities to observe an inspection, or obtain the results of an inspection carried out by another member
- Publishing an MDEP vendor inspection protocol document with guidelines for witnessed and joint inspections to facilitate inspections that are observed and attended by multiple regulators
- Completing 13 witnessed inspections, in which one regulator performs an inspection to its criteria, observed by representatives of other MDEP countries
- Completing an evaluation of the quality assurance requirements used in the oversight of vendors, including those areas where the various regulators have common regulatory frameworks
- Completing a comparison table of the ASME, RCCM, JSME and KEPIC codes for Class 1 pressure vessels
- Reaching agreement by the SDOs that they will work together to reduce additional divergence of the codes

- Drafting six common positions in the area of digital instrumentation and controls: software common cause failure, software tools, independent verification and validation, data communication, simplicity in design, and complex electronics
- Establishing a formal process to generate and process enquiries from member countries to promote an efficient and structured information exchange
- Issuing a paper on the 'Structure and application of high level safety goals'.

A5.6 Future trends

MDEP has begun to consider the addition of new topics and how they could be addressed by the programme. The criteria that will be used in evaluating whether an activity should be undertaken as part of MDEP include the following:

1. The activity is of generic interest and of safety significance to the licensing of new reactors in MDEP member countries.
2. The approach followed by the MDEP regulators is not completely similar.
3. Successful completion of the activity would likely result in increased harmonization/convergence in regulatory practices or increased cooperation within a reasonable timeframe and resource expenditures.
4. Any new MDEP activity should not duplicate similar efforts that are already ongoing or are planned to be undertaken by other organizations.
5. Each new activity should have a lead country willing to take an active leadership role, and should have a defined product.

In addition, a number of topics have been identified in which MDEP can play a significant, positive role by cooperating with current efforts in other organizations. Therefore, the MDEP STC will search out areas where it can act as a catalyst for enhanced regulatory cooperation and convergence in other forums. MDEP is in a unique position to effect positive change because it includes the regulatory authorities of over three-quarters of the reactors worldwide and represents those agencies at the highest levels.

A5.7 References

MDEP (2008), *Pilot Project and Assessment*, a MDEP Pilot Project report, OECD/NEA, Paris (www.oecd-nea.org/mdep/mdep_pilot_project_report.pdf)

MDEP (2010a), *Issue Specific Working Groups: General Terms of Reference,* OECD/NEA, Paris (www.oecd-nea.org/mdep/TOR-ISWG_Final.pdf)

MDEP (2010b), *Design Specific Working Groups: General Terms of Reference*, OECD/NEA, Paris (www.oecd-nea.org/mdep/ToR_DSWGs_Final.pdf)

MDEP (2010c), *Conference on New Reactor Design Activities 10–11 September 2009*, OECD/NEA, Paris (www.oecd-nea.org/mdep/events/conf_sept_2009/agenda.html)

MDEP (2010d), *Multinational Design Evaluation Programme: 2009 Annual Report*, OECD/NEA, Paris (www.oecd-nea.org/mdep/MDEP-Annual-Report-2009.pdf)

MDEP (2011), *Multinational Design Evaluation Programme: Terms of Reference*, OECD/NEA, Paris (www.nea.fr/mdep/mdep_TOR.pdf)

WNA (2010), *International Standardization of Nuclear Reactor Designs*, WNA, London (www.world-nuclear.org/)

1

Overview of infrastructure and methodologies for the justification of nuclear power programmes

A. ALONSO, Universidad Politécnica de Madrid, Spain

Abstract: Nuclear power has passed from a phase of euphoria to a long stagnation in many countries. The economic and environmental advantages of nuclear power have been recognized and a new phase of builds has started in new entrant countries and those with operating plants. In deploying nuclear energy the decision makers need to consider the long commitment, the concerns for safety and security, the high investment, the need for a scientific and technological infrastructure and the long-term management of radioactive waste. The full development of nuclear power requires the establishment of a global nuclear safety regimen based on national and international regulatory activities and on the justification of the desired programme.

Key words: nuclear power development, global nuclear safety regimen, nuclear power sustainability, used fuel management, nuclear power justification.

1.1 The past, current and future phases in the development of nuclear power

The development of nuclear power has passed through different phases. There have been countries, notably France, Japan and South Korea, among others, where nuclear development has been maintained steadily through time, while in other countries, notably in Western Europe and in the USA, the first phase of euphoria has been followed by a long period of stagnation. Currently governments in many developed and developing countries have been pondering about the need for nuclear power, and a new euphoria for nuclear power – some call it renaissance – is building up in a more mature and reasonable way than before. This part of the chapter describes the origins and development of nuclear power, as well as current interests in the field.

1.1.1 The beginning and the euphoria of the pioneers (1953–1979)

The historic speech addressed by President Eisenhower in 1953 to the United Nations General Assembly is considered the beginning of nuclear power development for peaceful purposes. It certainly aroused a general enthusiasm, first among the then nuclear countries and later on all over the world. In many opening meetings and inauguration ceremonies, as in Calder Hall, officers said and believed that electricity generated by nuclear power will be so abundant and cheap that it will not be necessary to meter it. This euphoria propagated rapidly around the globe and every country started to think about developing nuclear power to generate electricity.

This first phase of excitement started to decline in the early 1970s, and in the 1980s it was converted into despair. The first large, for the time, commercial nuclear power plants put into operation were not as cheap as it was assumed, the construction times started to grow longer, the capacity factors were not as high as expected, and in many countries subsidies were needed. Moreover, regulatory requirements became more strict and demanding, quality assurance, maintenance and in-service inspection needed advanced technologies that were not always available, and environmental radiological impacts and radioactive waste management were not conducted in the most effective ways. Moreover, first within the industry itself and later on within some social organizations, a strong nuclear phobia started to grow fast within society. In March 1979 the TMI-2 accident erased the primitive euphoria and caused the cancellation of many nuclear power projects, mainly in the USA.

1.1.2 Nuclear phobia and the stagnant phase (1980–2000)

Although the TMI-2 accident did not cause any relevant radiological consequences, it suddenly revealed the vulnerability of nuclear power plants. A never-envisaged core meltdown was possible due to a combination of equipment failure and human error. Moreover, the Governor of Pennsylvania ordered the evacuation of the most sensitive part of the population – children and pregnant women – based on the wrong advice from the Chairman of the Nuclear Regulatory Commission, NRC, that hydrogen explosion (impossible in practice, because of the absence of oxygen) within the pressure vessel could occur. These facts had a great impact on the growing nuclear phobia and on the lack of confidence in the industry and the regulator.

The many analyses conducted on the root causes and the development and consequences of the TMI-2 accident discovered the need for improvements in the design and operation of nuclear power plants and, more relevantly, the possibility of accidents producing severe damage in the core of the reactor. Regulatory requirements multiplied; new administrative procedures were promulgated; new instruments to cover severe accidents were required; and a relevant research programme to better know the phenomenology associated with severe accidents was soon initiated by the industry and the regulatory organizations. From that research effort, conducted within international participation, the science associated with severe accidents was understood and made it possible to develop technology to prevent severe accidents and to mitigate their consequences, which was incorporated, up to the maximum possible level, in the current reactors and is fully integrated into the new designs.

Nuclear phobia, enhanced by the TMI-2 accident, was a major factor behind the 1980 Swedish referendum which forced the government to establish a moratorium in the construction of new units and in fixing a programme of closing down the operating units by 2010, which has only partially been completed and is being reviewed. Social and political nuclear phobia, also enhanced by the TMI-2 accident, played a significant part in the decision taken in 1983 by the government of Spain to cancel the advanced construction of five large nuclear units and to establish a moratorium on the construction of new plants, which paralysed the expected development of nuclear energy in the country. Such phobia still exists in some political parties and non-governmental organizations, which are requesting the shutdown of the existing nuclear units, despite their high safety levels and recognized economic advantages.

Although of different design and with less strict operation requirements, the 1986 Chernobyl accident increased nuclear phobia all over the world, which produced a cancellation of nuclear projects, the conducting of referenda and the stagnation of nuclear development. Only a few countries, notably France, the Soviet Union, and some Eastern European and East Asian countries, continued with their nuclear development programmes.

The Chernobyl accident was behind the Italian 1987 referendum, which resulted in the complete disappearance of nuclear power installations in the country. Four nuclear units in operation and two under construction were cancelled. In its intention to renovate the nuclear fleet, the present Italian government has estimated that the cost of the decision to the country amounted to some 50 billion euros. The nuclear phobia shown in Germany against the transportation and storage in the country of high-level radioactive waste – from the reprocessing in France of German used fuel elements – was at the root of the country's coalition government decision in 2000 to establish a new nuclear law limiting the total power produced in the 17 German operating nuclear units and prohibiting the construction of new nuclear plants.

1.1.3 The current renewed interest in nuclear energy

The increase of greenhouse gases in the atmosphere, the depletion of gas and oil resources and the volatility of their prices, the intermittence, low efficiency and high prices of renewable sources of energy, and the economic advantages of nuclear power plants have all stimulated the worldwide interest for nuclear energy, which may start a new renaissance based on the improved designs of water-cooled reactors.

Since 2000, society has become increasingly aware of the potential impacts of climate change, in part induced by the emission of greenhouse gases coming from the combustion of fossil fuels used for electricity production. At the same time, society has started to realize that safety in the operation of nuclear power plants has been constantly improved and new safer nuclear plant designs have been developed by the reactor suppliers. Some of these new designs have also been submitted to a certification process by the US Nuclear Regulatory Commission (US NRC) and the UK Office for Nuclear Regulation (UK ONR), and there is international interest in harmonizing the new designs through international organizations, such as the Nuclear Energy Agency of the Organization for Economic Co-operation and Development (NEA/OECD) which is driving the Multinational Design Evaluation Programme (MDEP). The MDEP project is described in Appendix 5 of this book. All these factors form the basis of the new interest in nuclear power.

Although some countries in Western Europe still keep their moratoria on the construction of new nuclear power plants, other European countries, Finland and France in particular, have started the construction of new plants, while others, mainly the UK and Russia, have announced ambitious nuclear power programmes for the next two decades. The Asian countries, Japan, South Korea, China and India, that remained active during the stagnant phase have accelerated their nuclear power programmes. Likewise, American countries are also considering building new nuclear power plants, at a slower pace. All these ongoing activities, and many others not mentioned, sustain the idea that a new deployment of nuclear power is on the way.

The Japanese earthquake of 11 March 2011 and ensuing tsunami left the Fukushima Dai-ichi nuclear station without external power and an ultimate heat sink; despite the efforts made it was not possible to cool the reactors efficiently, and the reactor core in three of the six nuclear units in the site melted, releasing radioactive products to the atmosphere and the sea. These events have prompted the revision of the safety of currently operating nuclear power plants to test their abilities to cope with extraordinary circumstances. Such worldwide studies will serve to improve the safety of current and future nuclear reactor designs; nevertheless the events in

Fukushima have increased the social nuclear phobia and created a certain delay in the renewed interest in nuclear power.

It is foreseeable that the new deployment will be based on thermal reactors belonging to the so called Generation III+, fuelled with enriched uranium, and cooled and moderated by pressurized water (PWRs) or boiling water (BWRs). The advanced Canadian heavy water reactors (HWRs/CANDU) will also be built in a few countries. The useful lifetime of these new builds will be 60 years or longer, therefore such new generation will cover the largest part of electricity generation by nuclear power in the twenty-first century. There will be sufficient uranium for a reasonable deployment of such designs; in most countries the fuel cycle will remain open, but the used fuel will probably be stored for future reprocessing, needed to keep nuclear energy sustainable.

To achieve that sustainability it will be necessary to design, test and deploy the so-called Generation IV reactors. There are two international projects to that aim: the International Atomic Energy Agency (IAEA) driven INPRO project, with Russia the major sponsor, and the US Department of Energy (DOE) driven GIF project. Moreover, within the EURATOM framework research programmes there are several projects going on. The objective of this book is centred on current thermal reactors; the connexion with Generation IV reactors is only on the potential utilization of the fuel used in the thermal reactors.

1.2 The main factors shaping the deployment of nuclear power

The deployment of nuclear power is controlled by factors of different kinds that vary country by country. Some of these factors have to do with the decision makers, such as the recognition that nuclear energy implies a long-term commitment; there are also technical and economic limitations related, for instance, to the selection of the site, the technology to be deployed and the capital to be invested; there is also a concern for nuclear safety and security, non-proliferation and management of radioactive waste. The nuclear phobia strongly defended in many countries and society groups is also a hindrance to be considered. In the following paragraphs these issues are analysed.

Although many countries foresee rather large deployments of nuclear power plants, all of them are considering the factors mentioned above that shape such developments. New requirements from the analysis of the Fukushima events, mainly those related to siting and safety under extraordinary circumstances, will also be considered. The IAEA is also advising new countries on the need to judge the impact of such factors in the national development. A relevant ministerial conference was held in

Beijing in 2009 to consider the development of nuclear power in the twenty-first century (IAEA, 2009). The conference addressed the need for nuclear energy and its impacts. Plenary sessions were maintained to analyse energy resources and the environment; available technologies and the long-term perspectives; the need for developing infrastructure and a legal system to ensure safety; and waste management and strengthening non-proliferation.

1.2.1 The long-term commitment

It is widely recognized that embarking on a nuclear power programme, including the construction of a single nuclear power plant, implies a long-term commitment (more than a century) affecting many national administrative and industrial institutions. Table 1.1, based on the IAEA International Nuclear Safety Group (INSAG) publication INSAG-22 (2008a), defines and suggests the time-span taken by the different phases in the so-called *chronological lifetime* of a nuclear power plant, from the first steps down to returning the site back to greenfield status as defined by Tipping (2010).

After return to greenfield status, the low- and intermediate-level radioactive wastes, probably in a final repository, will maintain some type of monitoring; most importantly, the high-level radioactive waste coming from dismantling has to be stored in an intermediate facility before it goes to a final repository, probably in a deep geological facility. The used fuel may be stored in temporary facilities waiting to be recycled or considered as a high-level waste. In any case, the used fuel (if defined as a waste) or the waste coming from reprocessing has to be placed in the final repository. These activities, and the required monitoring, may extend the commitment to several additional centuries.

In the pre-decision phase (1 to 3 years) the major activities correspond to the planning authorities in the country, those in the administration and in the industry. In entrant countries, the governments have to enact or review basic legislation to cover the new nuclear activities and create, as soon as possible, a regulatory body. On its side, the industry has to establish a strong project management organization.

In the decision phase (3 to 7 years) a detailed nuclear power programme should have been developed by the government and approved by the legislature and possibly submitted to a public consultation process. In new entrant countries, the recently created regulatory organization will have to establish a licensing methodology and draft safety requirements for the coming site and construction permits and licences.

The implementation phase (7 to 10 years) is divided into two distinct sub-phases, site selection and construction. In those phases the govern-

Table 1.1 Main phases in the life of a nuclear power plant

Phase	Duration (years)	Major activities	Needed infrastructure
1. **Pre-decision**	1–3	Develop a nuclear plan. Develop basic legislation. Conduct a public consultation.	Establish a strong project management organization.
2. **Decision**	3–7	Develop a detail nuclear programme. Select a technology, a site and a supplier.	Create a nuclear regulatory body. Establish a licensing methodology. Develop safety requirements for siting and construction.
3. **Implementation**			
3a. Site selection and characterization	2–3	Characterize the selected site. Formulate a site licensing authorization.	Develop competence on earth sciences and man-made external inputs. Develop corresponding safety requirements and safety guides.
3b Design and construction	5–7	Initiate site preparation, equipment procurement and detail design. Fill application for the construction permit. Construct the plant in accordance with the safety requirements. Verify quality.	Develop national industry to optimize participation in design and construction. Develop competence on deterministic and probabilistic safety analysis, inspection during equipment fabrication and plant construction and on quality assurance.
4. **Operation**			
4a. Testing and commissioning	1–2	Verify that components, systems and structures comply with safety requirements. Apply for operation licence. Reach first criticality and perform established nuclear tests. Transfer knowledge and responsibility to operating organization.	Availability of well-trained operating personnel. Hold regulatory competence for the review and approval of test results and licensing reactor operators. Develop basic requirements, procedures, equipment and facilities to cover nuclear emergencies.

Table 1.1 Continued

Phase	Duration (years)	Major activities	Needed infrastructure
4b. Commercial operation	40–60	Operate the plant within the safety requirements. Establish a strong safety culture. Perform periodic safety reviews. Evaluate malfunctions, incidents and accidents. Prepare for refuelling outages. Conduct emergency drills.	Hold, maintain and increase regulatory competence on safety codes and standards for operation. Be part of the international global safety regimen and nets for safe operation and sharing operating experience. Develop competence on equipment ageing, radioactive waste and spent fuel management.
5. **End of life**			
5a. Decommissioning	5–10	File a dismantling plan for approval. Develop a radioactive waste management system. Enhance workers' internal and external dosimetry.	Develop safety requirements for decommissioning. Have a final repository for radioactive waste coming from operation and dismantling. Have a management programme for used fuel.
5b. Long-term management of spent fuel	15–100+	Establish and maintain long-term radiological control of used fuel and high-level waste.	Maintain competence on verification of compliance with regulations related to the long-term storage of used fuel and high-level waste.

Source: reproduced with permission from INSAG-22 (INSAG, 2008a).

ment's responsibilities are limited to overseeing the process and to responding to the regulatory body's reports in the way defined in the regulations. The regulatory body should be competent and prepared to evaluate the request, first for a site permit and later for a construction licence. These requests have to be properly documented by the applicant to demonstrate that the site and the proposed design and its construction and their interdependencies comply with the previously established requirements. For the first nuclear power plants, the concerned institutions will need to outsource the needed knowledge and experience, while maintaining their formal responsibilities.

The operation phase (40 to 60+ years) is also divided into two sub-phases. The testing and commissioning phase is a short – 1 to 2 years – but very relevant part in the life of the nuclear power plant. At the end of the process the plant is formally transferred from the supplier to the operator, who has to accept full responsibility for the safety of the plant. There should be a competent and sufficient group of well-trained persons to cover the multiple activities required to operate a nuclear power plant. The regulatory body needs to have sufficient knowledge and experience to decide that the verification tests have been conducted satisfactorily, that the results are acceptable and that the operating crews have sufficient knowledge and skills. For the first nuclear power plants, outsourcing such activities becomes a need; nevertheless the responsibility remains within the licensee.

During the long commercial operation phase (60+ years) the licensee has to operate the plant safely, reliably and economically, strictly keeping to the regulatory requirements for operation. The regulatory organization should develop a permanent and effective safety oversight system, capable of discovering departures from the establish requirements, hidden faults and tendencies and proper operation.

The end phase (8 to 10 years) starts when the plant can no longer comply with the safety requirements established by the regulatory organization – in most cases due to ageing – or its operation does not comply with the economic returns expected from the owner. After about three years of being shut down, when short-lived radioactive isotopes have decayed sufficiently, the dismantling of the plant may start. This process takes about five to six years for the site to come to a greenfield (or similarly suitable) status. During the latent period the plan owner has to keep the plant under control and prepare for decommissioning. This operation is normally done by a state-owned company, which becomes responsible for such operation. The regulatory organization has to grant a dismantling licence and regulate the classification of the radioactive materials to be removed or declassified. There could be very low activity, low and medium activity and high-level long-lived activity materials. Each class has a different management system.

1.2.2 Safety and security: a major concern

In countries with operating nuclear power plants, experience shows that any perceived slippage in safety, however small, would have an adverse impact on any existing favourable attitudes towards nuclear energy. It is also said that a reactor accident anywhere in the world will also have a negative impact everywhere. Likewise, security, i.e. the prevention of sabotage or the theft of nuclear materials, is also a major public concern, mostly in those countries where such acts might be more likely to occur. These public concerns about the safety and security of nuclear power plants have already been a barrier to nuclear power development and may also serve as a barrier to new construction in new entrant countries.

In the fission reaction radioactive nuclides are generated and accumulate in the fuel matrix. Close to 300 nuclides are born; some of them have a short life and soon disintegrate to stable nuclides, but others have longer lives. The excess neutrons also created in the fission reaction activate the fuel itself and generate radioactive actinides, some of them very long lived. The neutrons also activate the reactor core materials and the coolant, its impurities and additives. Some of these nuclides are very volatile, such as tritium, krypton-85 and iodine-131; others have an intermediate volatility, such as cesium-137 and strontium-90, among others; whilst others, such as plutonium-239, have a limited volatility and a long life. In case of severe accidents or sabotage, these nuclides can be released to the environment, as in the case of Chernobyl and Fukushima, and create a radiological problem.

Nuclear safety is the part of nuclear science and technology that is aimed at preventing accidents and reducing their effects in the very unlikely event of their occurrence. The possibility, however small, of an accident or sabotage is at the root of the public's apprehension towards nuclear energy. Chapter 10 of this book describes the basic safety principles. There are members of the public who do not question the reliability of nuclear power plants and have full confidence in the regulators and operators; others question whether it is possible to operate nuclear power plants safely and securely; there are also the more apprehensive who prefer to apply the precautionary principle and believe that in case of doubt it would be better not to have nuclear power.

All these peculiarities and their potential impact on society and the environment have been known and accepted since the very beginning. From the start, it was recognized that nuclear energy needed to be regulated; an independent regulator will oversee all activities and have the authority to correct any deviation from the established requirements. Despite this oversight, safety cannot be absolute and there remains a residual risk, which can be accepted by society only if the benefits of nuclear power greatly surpass any potential detriment coming from its deployment. In 2006, the IAEA in its Fundamental Safety Principles (IAEA, 2006) recognized the importance of such a long-established comparison and called it the Justification Principle, taking the idea from the International Commission on Radiological Protection (ICRP, 1990) when applied to the use of radiation, mainly for medical purposes.

The residual risk inherent in the operation of a nuclear power plant can be estimated by the so-called probabilistic methodology created in 1975 by a substantial body of US experts sponsored by the US Atomic Energy Commission (AEC), a precursor body to the NRC which then assumed nuclear regulatory functions. The so-called Reactor Safety Study (NRC, 1975) was made widely known and was applied in other countries. The methodology was further developed considerably; the nuclear risks were compared with those inherent in other industrial activities and natural phenomena to show that nuclear power plants are safer than most other industrial activities of comparable size. INSAG has considered this matter and has proposed (INSAG, 1992) that the expected frequency of an accident that may damage the reactor core should not be larger than 1 in 100,000 years of operation, and that the expected frequency of a large and rapid release of radioactivity should not be greater than 1 in a million years of operation.

Most of the currently operating nuclear power plants have undergone a specific probabilistic safety study, limited to level I – core damage expected frequency – and level II – expected frequency of radioactive releases. Although new designs also have a probabilistic safety analysis, it is recommended that any new reactor should undergo such analysis for the specific conditions of the site and the design. Both the licensee and the regulatory organization should develop knowledge and experience in performing and evaluating such safety studies.

Despite all these quantifications and limits, human beings, in their daily lives, do not quantify the many risks to which they are exposed; in a certain manner they develop a perception of risk and act in accordance with such perception. Although perception is an interior sensation coming from thinking and from the information transmitted by the senses and which is therefore subjective, this *perceived risk* is as real as quantitative risk estimations offered by the experts.

Currently, the perceived risk is very large in many countries and sectors of society, while the estimated risk is sufficiently small. This reality should be accepted and discussed among the parties concerned in order to take wise decisions without impairing the many social, economic and environmental benefits coming from nuclear power. In this discussion there are two additional risk concepts in favour of nuclear power: the management of risk and the compensation for damage.

In the defence-in-depth concept it has been widely accepted that the last line of defence should be based on the emergency plans. These plans, when well defined, equipped and managed, could limit the damage to the public and reduce the impact on the environment in the case of an accident. The existence of these emergency plans and the periodic testing of their efficiency have been established in all operating plants and should be incorporated in any new plant. In its Safety Standards Series the IAEA has provided requirements for nuclear or radiological emergencies (IAEA, 2002). Emergency planning is treated in Chapter 12 of this book.

The current revision of the Paris Convention on Nuclear Third Party Liability will allow for a considerable increase in the amount of compensation available to affected persons by a nuclear power plant and for expansion of the scope of the Convention to damages to the environment (NEA, 2004a). This amendment and similar ones have also been included in the Vienna Convention on Third Party Liability. These revisions should be considered as an additional guarantee for protection against any potential damage to the health and safety of the people and against damage to the environment.

1.2.3 Nuclear power is intensive in capital

Any investment in large-scale generation of electricity has to consider three major items: the so-called overnight cost (i.e. the capital that has to be invested to build the plant to which the interest rate has to be added), the fuel cost, and the operation and maintenance (O&M) cost. All these ingredients have to be manipulated to obtain the busbar cost of the electricity generated in the plant. The investment is the major component, while the cost of fuel is low and reliable, as well as the O&M cost.

Nuclear power plants are intensive in capital; i.e. the investment needed per unit of power to build a new nuclear power plant is very large, sometimes overriding the financial capabilities of small- and medium-sized electrical utilities. Even for large utilities the cost of a large nuclear power plant represents a significant fraction of the company's worth, and this large investment may put the company at risk should construction be delayed significantly or costs escalate appreciably. Nuclear power plants are at disadvantage in this respect and efforts are in place to guarantee that there will not be, as in the past, construction delays due to regulatory requirements or stakeholder interventions.

On the other hand, the cost of nuclear fuel is much less significant and more stable than the price of fossil fuels, which is a notable advantage for nuclear power plants. The O&M costs and other external costs are in all cases very much dependent on local conditions.

There are specific protocols for analysing the economy of any given plant and establishing comparisons with other sources for generating electricity. The results depend very much on the input data and on the financial practices and nature of the producer; moreover these evaluations are not generally available in the public domain. The NEA/OECD, in collaboration with the International Energy Agency (IEA), periodically conduct analysis of the projected costs of generating electricity, taking as a basis the data for real projects supplied to the OECD by the national authorities (OECD, 2011). Chapter 15 of this book considers the economics of nuclear power.

From these comparative analyses among large thermal power stations, it is traditionally found that the investment cost is highest for nuclear and therefore sensitive to the rate of interest, but the busbar cost of electricity is the cheapest for nuclear in all cases. In most cases, on average, the cost of the investment represents 70% of the total cost of the electricity produced, while the fuel contributes only 10% and O&M covers the remaining 20%.

Due to the high investment, financial markets will be wary of investing in new nuclear plants until it is demonstrated that they can be constructed on budget and on schedule. This is forcing utilities to require subsidies and financial guarantees to protect the investment. This has been granted by the US DoE for the first few plants, while the UK government has announced that there will not be any government support for new nuclear builds in the country.

In the past, one source of delays in construction came from changes in regulation. Chapter 20 addresses the licensing system; typically, licences are needed for the site, construction, commissioning and operation, and decommissioning. Once the work has started, it is expected that the process will go ahead to the next phase in the life of the plant. The US NRC has simplified the regulatory system by including early site permits, certified designs, and a combined construction and operation licence (COL). Other regulatory organizations are also streamlining the licensing process without losing effectiveness.

Another source of construction delays in the past has been associated with stakeholder court demands. This was particularly acute in the construction of the Sizewell B plant in the UK. Stakeholder involvement in the project has to take place before construction starts, as recommended by INSAG (2006a). The government should start a well-laid-out stakeholder consultation programme at a very early stage, and the decision to establish a new nuclear power programme should also have parliamentary agreement.

In the UK a formal justification process for nuclear designs, with the participation of stakeholders, has been put into practice. The justification authority has declared as justified the French European Pressurized Reactor (EPR) design and the Westinghouse AP-1000 design. This means that utilities applying to build one of these justified projects in a given accepted place will only have to inform and consider the opinion of the local people.

Any new nuclear power plant build should only start when there are enough guarantees that construction will not be delayed due to unnecessary regulatory changes, stakeholder formal involvement or other causes.

1.2.4 Nuclear power is intensive in intellect

Nuclear physics is the scientific support for nuclear technology; it has been developed throughout the twentieth century and is still developing. It is a complex science requiring advanced mathematics and abstractions that are often difficult to conceive. The physical phenomena taking place in the core of a nuclear reactor are much more difficult to understand than those present in the combustion chamber of a fossil fuel plant or those affecting the operation of a wind turbine. Reactivity control, a phenomenon unique to the reactor, and the transport of heat from the core to the energy conversion system also need the understanding of complex and interlinked phenomena, even under normal operating conditions, and much more during transient and accident situations. Neutrons also interact with the different materials in the core, changing their mechanical properties and producing activated materials. In nuclear technology, materials science is also in high demand, as well as quality assurance in the design and construction of structures, systems and components and in the operation of the nuclear plant. Quality assurance during design, construction and operation is considered in Chapter 21 of this book. These characteristics make nuclear technology intensive in intellect, with the result that the deployment of nuclear power plants requires a large number of well-qualified experts for the design, construction, operation and decommissioning of nuclear power plants.

Nuclear technology has created new materials, perfected technologies and technological processes, and developed strict codes and standards for the design, manufacturing and quality assurance of components. A unique industry has been created to produce nuclear power plant materials and components, which may not be sufficient to cope with a high demand. Moreover, the strict requirements concerning materials and fabrication techniques will limit the number of countries participating in the design and construction of nuclear power plants.

This intensity in intellect requires that all countries wishing to introduce nuclear power programmes need to have a scientific and technological structure to take an active part in the design, manufacturing of components, assembly and construction of the power plants. This requires the development of research and education institutions, as described in Chapter 7 of this book on the need for technological development. Although there could be other arrangements, such as outsourcing, the commissioning and operation of the plants have to be conducted under the responsibility of the licensee and this requires that the operating team has to receive and maintain a high level of education, as described in Chapter 6 of this book on the need for human resources.

International organizations, mainly the IAEA, the European Union and the NEA/OECD, have shown concern on nuclear education and training. During the IAEA 2002 General Conference, resolution GC(46)/RES/11B on nuclear knowledge was approved, and this resolution was reiterated in the subsequent General Conferences.[1]

One of the results of such a resolution was the creation in 2004 of the Asian Network for Education in Nuclear Technology (ANENT) to assist countries in the Asian region to build capacity, develop human resources, and construct scientific infrastructures through cooperation in education, nuclear knowledge management and related research and training. Currently there are many member institutions from 17 countries and several international and regional networks as collaborating members. ANENT is very active and can be reached through http://anent-iaea.org. A summary of the IAEA activities and international coordination with ANENT can be found in IAEA (2007).

The 'European Nuclear Engineering Network' project was launched under the 5th European Commission framework programme in January 2002. It established the basis for conserving nuclear knowledge and expertise, developing high-level nuclear education within the European Union member countries and establishing links within the learning centres and the end users of knowledge and expertise. In September 2003 the European Nuclear Education Network (ENEN) Association was established by the partners of the European Nuclear Engineering Network project. Because the project itself was limited in time, the foundation of the ENEN Association, a non-profit international organization under the French law of 1901, gave a more permanent character and a legal status to the foreseen activities. As of 11 March 2010, ENEN has 56 members, mainly European universities and some research centres. ANENT has established a partnership with ENEN. The many activities of ENEN can be found in http://www.enen-assoc.org.

The NEA/OECD has also shown serious concerns on the need for education and training in nuclear science and technology. In a reflection paper prepared in 2000 by the Working Group on Nuclear Education, Training

[1] In the IAEA General Assembly, proposed resolutions are either approved or denied.

and Competence (NEA, 2000) it is recognized that 'there are indicators that future (nuclear) expertise is at risk'. It also states 'The ability of universities to attract top quality students, meet future staffing requirements of the nuclear industry, and conduct leading-edge research is becoming seriously compromised'. In the follow-up document on nuclear competence building (NEA, 2004b), mechanisms and policies are identified for promoting international collaboration in the area of nuclear education and research and development. Concerns for losing expertise for research and development in nuclear safety have also been expressed by the INSAG (INSAG, 2003).

Responsible officers from the nuclear industry have also expressed their concerns on the potential lack of sufficient human resources for a future deployment of nuclear power plants. With the objective of changing that situation, the World Nuclear Association (WNA) is supporting the World Nuclear University (WNU). The World Nuclear University has been defined as 'a global partnership committed to enhancing international education and leadership in the peaceful applications of nuclear science and technology'. That partnership includes the WNA and the World Association of Nuclear Operators (WANO), has the support of the IAEA and the NEA/OECD and includes, as members, leading learning institutions in some 30 countries. As a symbol, the WNU was inaugurated in 2003 in a London ceremony commemorating the 50th anniversary of President Eisenhower's historic 'Atoms for Peace' initiative. WNU is a non-profit corporation pursuing its educational and leadership-building mission through different programmes, one of the most salient examples of which is the WNU Summer Institute at Oxford University.

High-level nuclear education alone is not sufficient. The construction and operation of a nuclear power plant also needs a large number of persons with practical knowledge in many activities such as welding, installing and fitting mechanical, electrical and electronic components of many kinds. This knowledge is normally acquired in vocational schools – in essence, the needed knowledge is no different from that required in other types of installations, but the nuclear application is generally stricter than in other industries and the work in the operating plant often has to be done under radiation. All this requires additional and deeper knowledge. This extra training is generally done on the plant premises under well-defined training programmes. The US Institute of Nuclear Power Operations (INPO), created in 1982, has developed training requirements for operations, maintenance and technical training programmes for key functional areas in each individual plant. The National Academy for Nuclear Training was established by the US utilities to 'focus and unify industry efforts to continue improvement on training and qualification programmes and to enhance professionalism and pride of nuclear personnel'. INPO has provided guidance on how to conduct effective training and the Academy has established the way to formally qualify the training programmes (ACAD, 2002). These guidelines for the conduct of training and the qualification of the training programmes have been accepted in other countries.

The remarks so far do not address public education on nuclear issues. In countries with nuclear power plants in operation, these plants are part of everyday life and often produce news of a different character which should be clearly explained in order to be better understood by the people. Even in countries without operating nuclear power plants, the use of radiation for medical or other purposes also creates news and is of concern to society. Any new entrant needs to be prepared to address the anxiety that a new nuclear power plant may create among the population. There are many countries where a strong antinuclear social opinion has prevented the development of a nuclear power programme. There is also a group of countries, Austria, Italy and Spain, for example, where plants already in operation, built or in an advanced state of construction have been stopped because of society's attitudes.

One of the key elements in overcoming antinuclear social attitudes rests on the creation of a national system of public education on nuclear matters along the educational chain. These activities have been going on for a long time in countries with operating plants. For example, in Spain, the Atomic Industrial Forum has maintained consistently, for many years, an information programme for science teachers in high schools. All nuclear power plants have built an information centre open to the public. Even the Spanish regulator has built an information centre covering its activities, and the national company for radioactive waste management also has such a facility. Similar examples can be found in other countries. It is also relevant that people working in the information media should have a minimum level of nuclear education to ensure precision when communicating nuclear news. The Spanish Atomic Industrial Forum has created a nuclear energy manual for journalists. There are also examples in other countries.

The demand for nuclear knowledge is very high at all levels and is difficult to acquire. The extreme antinuclear attitude of some sectors of society and their national and international influence in political decision making has already prevented the deployment of nuclear energy in many countries and is therefore depriving those countries of the social, economic and environmental advantages of nuclear power. The concerns of society should be addressed, but a proper understanding of nuclear issues needs to be based on a solid system of education and information.

1.2.5 Nuclear power is intensive in energy

The fission of a uranium-235 atom generates 210 mega electron-volts, MeV, of energy, while the combustion of an atom of carbon-12 produces only about 4 MeV of energy. Therefore, the fission reaction is some 50 million times more energetic than the combustion reaction. This is a significant difference; to generate the same amount of electricity a conventional power station will need millions of times more fossil fuel than the amount of fuel for a nuclear plant. This explains the low impact of the nuclear fuel in the cost of electricity and the relatively small amount of used fuel which is produced.

In the current design of light water reactors (LWRs), uranium is introduced in the reactor as an oxide, with uranium-235 enrichment that may vary from about 3 to 5%, only a fraction of which is consumed in the reactor. During the process, a small fraction of the uranium-238 present is transmuted to plutonium-239, which is also fissionable. The final result is that the mass of nuclear fuel per unit of energy generated is 2 to 3 million times smaller than that of coal, depending on the quality of the coal, and 1.2 million times smaller than natural gas. This reduced demand on natural uranium resources makes nuclear power more attractive.

Although uranium is a relatively common element, the mass of the natural resources of uranium which may be recovered at reasonable prices is millions of times less than those of coal and natural gas and some concerns have been expressed on the availability of uranium to cope with a large deployment of new nuclear power. In any case, the supply of natural uranium has the potential to introduce big changes in the future development of nuclear technology.

The so-called *Red Books* (named after their colourful cover), published roughly every two years since 1965, are one of the most reliable sources of information on uranium supply and demand. Originally those books were developed by an expert group within NEA/OECD. Since the mid-1980s the books have been developed in cooperation with the IAEA. In 2006, a compilation edition recreated the 40 years of work (NEA, 2006). The last Red Book was published in 2010 to cover 2008–2009 (NEA, 2010). It discusses the efforts being made to boost production to cope with the increasing demands. It also features projections on uranium requirements through 2035 as well as an analysis of long-term uranium supply and demand.

In a recent NEA outlook for nuclear energy development (NEA, 2008) it is stated that the 'current identified conventional uranium resources are already sufficient to fuel the NEA high scenario expansion of global nuclear generating capacity employing a once-through fuel cycle until 2050'. The NEA high scenario assumes that an average of 12 new reactors are built per year from 2007 to 2030, needed to replace decommissioning plants and increase nuclear power contribution in the generation of electricity. From 2030 to 2050 the building rate is projected to increase, reaching an average of 50 new builds per year, up to a final capacity three times larger than the current one. The growth of the uranium market may lend to the discovery and exploitation of further conventional resources, which may prolong the supply of uranium to about 100 years for the high-level scenario previously discussed.

Although there are still unconventional resources with a large potential capacity, a natural resource assumed to last for a century cannot be considered as a long-term solution for generating electricity. The once-through fuel cycle is a very ineffective way to use uranium. Reprocessing the used fuel will separate sufficient plutonium to be used in the same LWRs using MOX fuels, to increase the life of the resource by some 20%.

To improve sustainability of supply it would be necessary to deploy an appropriate combination of thermal and fast breeder reactors working with the uranium-238/plutonium-239 cycle. In the future, the thorium-232/uranium-233 cycle, under development in India, will also be used. In this way the complete energy capability of the natural uranium resource will be materialized, potentially increasing the life of the resource to several millennia, securing the future sustainability of nuclear energy for electricity generation and other uses. The reprocessing technology is commercially available and the fast breeder technology is well advanced. The deployment of this technology will be one of the most important nuclear power developments.

1.2.6 The need to control nuclear materials

It is well recognized that uranium-235 and plutonium-239 are strategic materials of great relevance to nuclear weapons programmes. The Non-Proliferation Treaty (NPT) was created within the United Nations with the intention of preventing the development of nuclear weapons. Scientists have been able to develop nuclear fuel cycles with the potential to prevent nuclear proliferation, but they are not yet commercially available. Of course, the best non-proliferation system rests on the willingness of decision makers in only pursuing peaceful uses of nuclear power; such willingness should be based on the principles of equality and universal solidarity.

Nuclear proliferation has a deep influence on the development of nuclear power. So far, the accountability of nuclear materials in declared installations has proven to be reliable and offers guarantees on the peaceful use of nuclear energy in those countries which have signed the IAEA Comprehensive Safeguards Agreement (CSA) system of accounting (IAEA, 1972) approved by the IAEA in 1972; it provides the basis for negotiating CSAs between the IAEA and NPT parties as required by Article III.1 of the Treaty. After the Iraqi experience (see also Chapter 13), the IAEA proposed a Model Protocol Additional to Agreements for the Application of Safeguards (IAEA, 1997) to improve the effectiveness of the system, based on facilitating the accountability of all nuclear materials and activities related to the nuclear fuel cycle and to have complementary access to locations additional to those defined in the CSA already signed. This subject is developed in Chapter 13 of this book.

So far, the most significant drawback to nuclear development attributable to the generation and control of strategic nuclear materials was the decision taken by US President Carter on 10 March 1978 to sign the Nuclear Non-Proliferation Act, to ban used fuel reprocessing and to declare that the nuclear fuel cycle had to be open.[2] This decision was not followed by France where fuel reprocessing has continued, covering not only its own fuel but also the fuel from other countries. This activity has recovered plutonium-239 which has been reused in the light water reactors in MOX-type fuel elements composed of a mixture of uranium and plutonium oxides, which therefore extends the country's uranium resources. The use of MOX fuel is practised regularly in France, the USA, Japan and other countries and is considered an advantage, as plutonium partially substitutes for uranium-235.

The maintenance of an open nuclear fuel cycle decision can have two negative impacts on the future of nuclear development. First of all, the used fuel would be considered as highly active radioactive waste, for which a final destination has to be found, most probably a deep geological repository. Secondly, the residual usable fuels, i.e. the unburnt uranium and the generated plutonium, would not be used to generate additional nuclear energy, and this would in turn prevent the development of a sustainable nuclear energy system utilizing Generation IV reactors.

New entrant countries need to consider the destination of their used fuel. There are several solutions to this. In the near term, used nuclear fuel is stored in the *ad hoc* pools within the plant premises. Generally the capacity of these pools will not cover all the used fuel for the life of a plant. This should not be considered an impediment to a nuclear power programme, as analyses and practical experience have shown that used fuel can be stored for up to a century in dry cask storage at a very low risk of release of radioactive material to the environment. But this kind of storage is temporary and sometime in the future the used fuel will have to follow either the open cycle with storage in a deep geological repository, or the closed cycle with reprocessing. There is also an intermediate solution for entrant countries – that of leasing nuclear fuel from other countries which can then be sent back to the provider once it has been used. Within the IAEA there have also been suggestions that would internationalize the fuel cycle, i.e. with enrichment and reprocessing conducted only in specific international

[2] President Carter convened a large group of international experts who concluded that the fuel cycle of the moment was not proliferation proof, after which he signed the Act and ordered the cancellation of the almost finished Barnwell reprocessing plant.

facilities under strict control. Nevertheless, the materialization of these plans has not advanced much and there exists the possibility that they will not materialize at all, at least on a global scale. Even less advanced is the development of an international repository for used fuel for the highly radioactive waste arising from reprocessing and for active in-core components resulting from dismantling.

New entrants should consider the convenience of developing their fuel cycle supply and used fuel management systems. The International Advisory Board (IAB) chaired by Hans Blix, former IAEA Director General, set up by the government of the United Arab Emirates (UAE), has recently published its first semi-annual report (IAB, 2010) where it recommended that: 'The UAE programme accelerate the development of an integrated fuel cycle strategy including methods for securing long term supplies as well as arrangements for the management . . . of spent fuel and other nuclear waste.' This recommendation is also valid for any new entrant to nuclear power and to countries with operating nuclear power plants that have not yet established a final policy for the management of used fuel.

1.2.7 The long-term management of nuclear waste

In many Western European countries citizens do not support nuclear energy because of the wide belief that the management of nuclear waste is not adequately solved. In Germany, for instance, the biggest ongoing antinuclear demonstrations have been focused on the transportation by rail of vitrified high-activity waste arriving from the French La Hague reprocessing plant to the German Gorleben repository. Similar demonstrations were seen in Barcelona during the transportation by rail through the city of irradiated fuel to France from the Vandellós nuclear power plant. It is also believed that the termination of the USA Yucca Mountain project for the storage of used fuel was not due to technical reasons but because of social and political pressures. The management of radioactive waste, mainly the used fuel, has been and will continue to be a major drawback for nuclear deployment, if not approached properly.

The management of radioactive waste has passed through three distinct phases, each driven chronologically by the pioneers, the managers and the scientists. The first one has already passed, while the last two currently coexist. The management of radioactive waste is considered in Chapter 14 of this book.

Although radioactive waste management has always been a matter of concern, nuclear pioneers did not put sufficient attention into achieving proper management in the early days, where the major concerns of scientists, technologist and industrialists were concentrated on research and development into designing, building and deploying nuclear power plants and fuel cycle installations. Radioactive waste from research was introduced into drums and buried in shallow trenches. For a time, US drums containing radioactivity were dumped into the Pacific Ocean, and European wastes (under the supervision of the NEA/OECD) were dumped into the Finisterre depression in the North Atlantic Ocean, not far from the Spanish coast. These practices were considered unacceptable and terminated in the 1980s.

Following the pioneering closure of the fuel cycle, the logical solution was to reprocess used fuel, mainly to recuperate the generated plunonium-239 of strategic value at that time. For a while, high-level liquid wastes coming from reprocessing plants were dumped into the sea; but again, it was soon recognized that such practices were not acceptable and new management systems were developed leading to the vitrification of the liquid wastes and the storage of vitreous materials on land.

The first management activities were not convincing and, once they became more widely known, were not accepted by society. It was clear that a new managerial system was needed. Countries with nuclear power plants in operation created specific institutions to manage radioactive waste and used fuel. National institutions very soon started to develop and introduced radioactive waste management activites and solutions, such as ANDRA (France), DDA (UK), DBE (Germany), COVRA (The Netherlands), ENRESA (Spain), NUMO (Japan), NWMO (Canada), OCRWM (USA), POSIVA (Finland) and SKB (Sweden), among others, as well as some other multinational institutions, such as EDRAM and COWAM. They soon came to the idea that the best solution was to build above-ground repositories for low and medium activity wastes including radionuclides with short and intermediate lifetimes. For high-activity, long-life wastes deep geological disposal was foreseen as the best solution. Long-term research programmes were also soon established nationally and on cooperative bases internationally.

Nevertheless, many experts believed that such decisions for waste management strategies, using mainly deep geological repositories, were not based on solid scientific knowledge and a proven technological basis. The decision to develop deep geological repositories was mainly based on (1) the perceived urgent need to solve the problem, and the estimated possibility of developing such repositories with the current mining technology and with a minimum of research efforts; (2) the desire in some countries during the 1980s and 1990s to phase out nuclear power plants completely, and (3) the belief that the repository solution was the best to protect future generations from the radioactive wastes generated by the present generation.

This solution, as stated above, has not always been perceived to be acceptable. The continuing resistance to the Gorleben repository in Germany, the Yucca Mountain Project in the USA and the Berrocal project in Spain, among other examples, clearly proves that actions have to be taken by the governments to accommodate these concerns. INSAG has recommended creating a system of information and participation of stakeholders in relevant nuclear activities, including used fuel management (INSAG, 2006a). The US Nuclear Waste Policy Act of 1982, amended in 1987, by giving some oversight responsibilities to the local governments has improved the situation in the US. In Europe some countries, notably Finland and Sweden, have developed a strong, longstanding, well laid out and very successful information system. More recently, EU leaders have created the European Nuclear Energy Forum (ENEF) to create an open dialogue with stakeholders.

Apart from the social reaction, the experts recognized the difficulties in engineering and proving the validity of a deep geological repository. The managers themselves were aware of such difficulties and soon introduced two new concepts: reversibility and retrievability. Reversibility means that the process could be stopped in any of its phases, if needed. Retrievability assumes that the waste could be retrieved, completely or in part, in case the contents, the repository, or both, did not behave in the expected manner. It could prove impossible, or very difficult, to retrieve the fuel after a repository has been sealed.

A new scientific approximation could be developed if it were possible to increase the decay constant of the radionuclides in the waste.[3] The decay process is statistical in nature and it has not yet been possible to modify the decay constant in any physically available way. Marie and Pierre Curie, after discovering radium and measuring its decay rate under normal laboratory conditions, repeated such measurements with the material subjected to very low temperatures and to very high pressures, and did not find any difference in the measured rate of decay. Their experiments proved the invariability of the decay constant under the tested environmental conditions, but these findings do not mean that there could be other physical agents, energy fields or circumstances, not yet known, that could modify the rate of disintegration.

Transmuting radionuclides by neutrons is a well-known physical process. These transmutations may produce shorter-lived radionuclides and even convert radionuclides into non-radioactive isotopes. This idea, first proposed by the Italian Nobel Prize winner, Dr Carlo Rubbia, has been considered in several research projects. To achieve transmutation it is necessary to use large fluxes of high-energy neutrons, which can be obtained in Accelerator Driven Spallation facilities and fast reactors. The application of such processes to manage used fuel requires the chemical separation of the long-lived actinides and fission products followed by irradiation by a

[3] This is an idea coming from some observations in the Sun. It is believed that the decay constant of some radioactive products in the Sun may be shorter than in the Earth due to the fields there, though this is currently very speculative.

neutron source with the indicated properties. The complete process is called separation–transmutation or partitioning and transmutation. Well-founded research projects are under development in this area, mainly sponsored by EURATOM.

Although many processes are being explored, there is not yet a unique satisfactory solution.[4] The most promising solution is the separation–transmutation process. It has the advantage of recycling the used fuel and promotes the sustainability of nuclear fission power production. High-level radioactive products will still be produced by this process, but these wastes will have a much shorter life and they can be incorporated into glass, a stable solid matrix (i.e. vitrified). The technical and economic viability of the separation–transmutation process will need some time to be fully demonstrated; in the meantime, the used fuel can be safely stored in dry containers.

1.3 The bases for the development of nuclear power

The current recognized interest for developing nuclear power in new entrant countries and also in countries with operating nuclear power plants is based on the perceived advantages of nuclear energy to generate electricity. The demand for electricity and for other forms of energy, such as transportation in electrically driven vehicles and the production of high-temperature steam for the production of hydrogen and other uses, has created a need for nuclear power. It is highly recommended that the coming deployment of nuclear power plants is conducted within the global nuclear safety and security regimen defined by the IAEA. This means the early establishment of a satisfactory set of regulations based on the IAEA Safety Standards Series. The introduction of nuclear power in a country should also be justified by comparing the benefits to be obtained with the detriments it may create. These aspects are considered in the following paragraphs.

1.3.1 The need for nuclear power

The need for nuclear power is clearly demonstrated by the activities going on and the proposals already formulated by the IAEA Member States. The IAEA Director General in his address to the Board of Governors in the 2010 General Conference (IAEA, 2010a) announced that there were 441 nuclear power plants operating in 29 countries, with a total capacity of

[4] Transmutation is a way of reducing activity in the long range by changing long-lived nuclides into shorter-lived nuclides. After transmutation, activity increases but it disappears faster through decay.

375 GW(e) representing more than 15 millennia of operating experience. He added that a total of 65 reactors were under construction with a total electrical capacity of 62.9 GWe. It is of interest to note that by the end of 2009, 123 nuclear power reactors have been shut down permanently, of which 12 plants have been totally dismantled, 54 are in the process of being dismantled, 48 are kept in a secure enclosure, three have been entombed and six are pending a dismantling strategy.

The largest activity is going on within the 29 countries that already have operating nuclear power plants. Of these 29 countries, 13 are already building new plants, mainly China (27, including two in Taiwan), the Russian Federation (11), India (5) and the Republic of Korea (5). Four countries – Bulgaria, Japan, the Slovak Republic and Ukraine – are building two plants each, while five more – Argentina, Brazil, Finland, France and the USA – are each building one plant. Two countries with operating plants plan to phase out or limit the production of nuclear generated electricity, while five are reviewing their national energy plans, four are permitting new builds but without government support, and five countries will build power plants with government support.

The IAEA maintains a Status and Prospects on Nuclear Power information portal based on the information provided by the Member Countries. Currently more than 60 newcomer countries have expressed interest in developing nuclear power; although half of those countries are only interested in knowing more about the issues associated with nuclear power, the other half have shown the intention or have advanced plans to proceed with nuclear power development. The United Arab Emirates (UAE) has accepted a $20 billion bid from a South Korean consortium to build four commercial nuclear power reactors, totalling 5.6 GWe, by 2020; Bangladesh, Egypt and Vietnam have been planning for nuclear power for some time and Poland is reviewing the nuclear option. Up to 10 newcomers are expected to start nuclear power plans within a few decades.

Among the countries with already operating nuclear power plants, the US could be the most active. It has been published that as of July 2009, the US Nuclear Regulatory Commission (US NRC) had received 17 applications for combined construction and operating licences for 26 units, and it expects to receive a total of 22 applications for 33 units by the end of 2010. In early 2011, some 17 companies and consortia are considering building more than 30 nuclear power plants. The USNRC is actively reviewing 12 combined licence applications from 11 companies and consortia for 20 nuclear power plants.

Plans for new builds are very advanced in the UK. EDF Energy is planning to build two EPR units at the Hinkley Point site where two advanced gas-cooled reactors (AGRs) are in operation. Recently, the Dutch government has outlined the requirements for the construction of new nuclear power plants starting before 2015. The Czech Republic, Poland and Lithuania have announced plans to start new builds, while Slovakia and Romania are already building new nuclear plants. All these activities clearly demonstrate the perception in many countries of the need for nuclear power and its deployment.

The 11 March 2011 Fukushima events, although still pending from a thorough international analysis, are already affecting the pace of these ambitious projects. Countries with operating or under-construction nuclear power plants and those with advanced new nuclear power projects need to know in detail the Japanese experience to increase the safety of their nuclear units and to introduce new siting and design requirements for any new build.

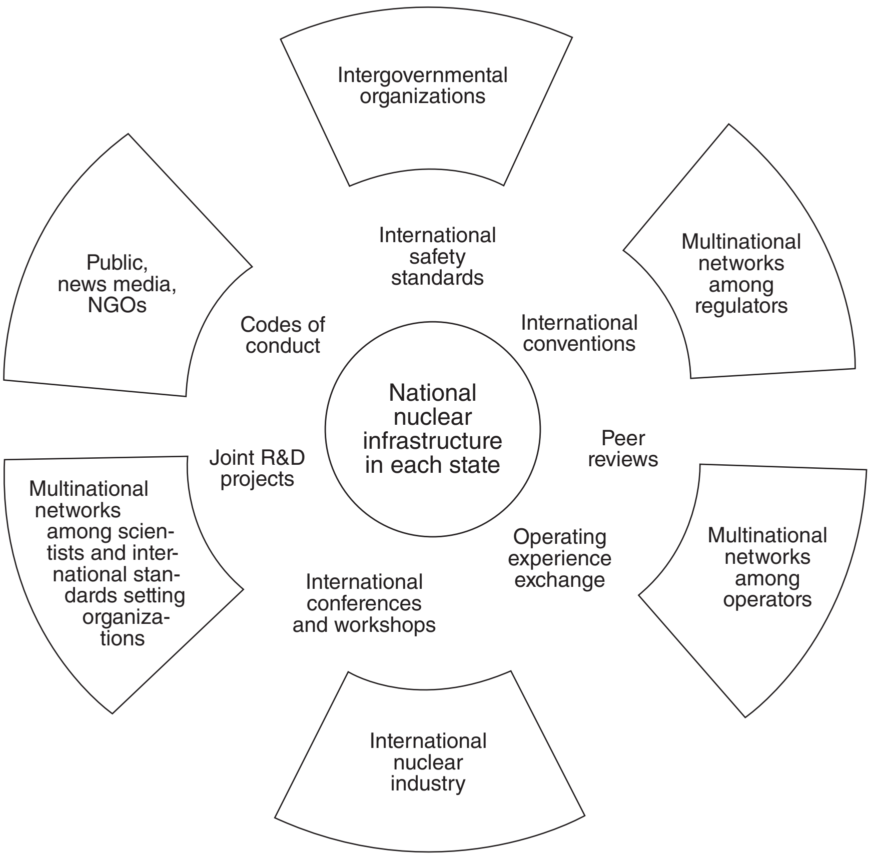
1.3.2 The need for a global nuclear safety and security regimen

The 1979 TMI-2 accident made very clear the need for a global regimen to share nuclear power plant operating experience. The subsequent creation in the US of the Institute for Nuclear Power Operation (INPO) was a clear response to such need. Its wide open invitation for participation of electric utilities in other countries initiated the creation of a truly global regimen on sharing operating experience. The 1986 Chernobyl accident also made very clear the necessity to involve all countries as active partners in a single global nuclear safety regime. Leading countries promoted the creation of several international conventions relevant to nuclear safety.

The increasing rate of terrorist activities has also led national authorities and international organizations, mainly the IAEA, to increase their security requirements to protect nuclear power plants against such acts. The IAEA International Nuclear Safety Group has specifically considered the need for strengthening the global nuclear safety regime (INSAG, 2006b). At the IAEA 2002 General Conference, resolution GC(46)/RES/13 was adopted to coordinate the IAEA security activities.

National regulators and power plant operators and vendors have created associations to harmonize existing regulations and develop common safety regulations and practices. The Western European Nuclear Regulators Association (WENRA) is one of the most active networks. The World Association of Nuclear Operators (WANO), created after the Chernobyl accident, is truly a worldwide network interchanging operational experience and safety practices. The MDEP, already mentioned in Section 1.2.3, is an outstanding effort to create uniformity in the safety requirements and design basis of new projects.

The main elements of a nuclear safety regime are depicted in Fig. 1.1 taken from the INSAG-21 document (INSAG, 2006b). There are three distinct levels in the wheel: the ‘motor’, i.e. core national infrastructure; the



1.1 The wheel of the global safety regimen. Reproduced with permission from INSAG-21 (INSAG, 2006b).

'transmission', i.e. standards, projects and cooperative tools; and the external organizations needed to achieve a truly global safety regimen. In its report, INSAG considers that the motor of the global nuclear safety regimen starts within the states with nuclear power plants under consideration, construction or operation. There are instruments in place to harmonize the national infrastructures, the most significant being the international conventions, mainly the Convention on Nuclear Safety (IAEA, 1994), the IAEA safety standards, considered in Chapter 4 of this book, and the interchange of operating experience, the importance of which has also been considered by INSAG (2008b). Finally there should be international organizations and multinational networks able to accept and develop the necessary harmonization of safety principles.

New entrants and also all countries with operating nuclear power plants should recognize the relevance of their contribution to the global nuclear safety regimen. Being a party to the relevant conventions, adopting a sound and complete set of regulations and participating actively in multinational organizations and networks are highly recommended.

INSAG recognizes that most of the elements in the wheel of the global nuclear safety regimen are already in place and working in the intended way. Nevertheless it encourages (INSAG, 2006b, page 8):

- 'Use of the review meetings of the CNS (Convention on Nuclear Safety) as a vehicle for open and critical peer review and a source for learning from the best practices of others'
- 'Enhanced utilization of the IAEA Safety Standards for the harmonization of national safety regulations to the degree possible'
- 'Enhanced exchange of operating experience and the use of this experience for life cycle management and back fitting of nuclear facilities, as well as for improving operating and regulatory practices'
- 'Multinational cooperation for the safety review of new nuclear power plant designs.'

Although security has been a requisite from the start of nuclear power development, it has remained within the plant itself, the national regulatory organizations and the government institution for civil security and protection. The details of any plant security programme have to be kept secret. The nature of security makes it non-amenable to a global security system. Moreover, the thread highly depends on the national conditions. Nevertheless, as it has already said, the IAEA has started to reorganize its functions on security. Since 2002, the Nuclear Safety Department has been converted into the Nuclear Safety and Security Department and some safety guides have started to appear (IAEA, 2008a, 2008b, 2008c).

1.3.3 The need for a complete and satisfactory set of regulations

Nuclear power has the potential to provide benefits to countries as a whole, society at large, individuals and the environment. But the parallel unavoidable generation of radioactive nuclides represents a risk to the society, the individuals and the environment. To reduce the risks to an acceptable level it is necessary to closely regulate the development of nuclear power. Chapter 4 of this book describes the regulatory requirements to be implemented to enhance safety and therefore to reduce the risks, while Chapter 20 describes the licensing process necessary to have the formal approval of the plant and to define the responsibilities for its operation.

Principle 2 of the IAEA Safety Fundamentals (IAEA, 2006) addresses the role of government. It established that 'An effective legal and governmental framework for safety, including an independent regulatory body, must be established and sustained'. It adds that such regulatory body should comply with three major activities: (1) to propose and enact safety requirements and safety guides that properly cover all nuclear activities foreseen in the country, including siting, design and construction, commissioning, operation and dismantling of nuclear power plants and fuel cycle installations, as well as the safe transport of radioactive materials and nuclear fuel; (2) to verify compliance with the licence conditions and with applicable regulations and to assess the safety of the installations and corresponding activities through safety analysis, oversight activities and inspections; and (3) to enforce the application of the established licence conditions and applicable regulations in case of departure or intentional deviation from such requirements and regulations.

The building up of a satisfactory and complete set of regulations is a long-term job that matures along with the development of nuclear power in a given country. A basic nuclear law should be proposed by the government and properly approved well before the decision is taken of whether or not to develop a nuclear programme. Such law should be in accordance with the idiosyncrasy and the legal structure of the country but, at the same time, it should include the basic principles of nuclear regulation, i.e. the protection of society, the individual and the environment against radiation risk. A second key element is the creation of the independent regulatory body, including its composition and functions, as well as means to enhance its independence, competence and human resources. The IAEA Safety Standards Series includes valuable documents to help countries in developing such documents (IAEA, 2010b).

Once created, the regulatory body has to grow in experience and competence. INSAG has considered such development on the basis of the IAEA Fundamental Safety Principles (INSAG, 2008a). The first major activity to be accomplished is the site evaluation and its compatibility with the design of the plant and emergency plan effectiveness. Detailed requirements and guides for site evaluations are available in the IAEA Safety Standards Series (IAEA, 2003) and presented and developed in Chapter 18. The main site characteristics of interest include population distribution; external natural hazards, mainly seismology, extreme meteorological conditions and hydrological events; human-induced events such as fires; and the possibilities of large chemical explosions nearby or aircraft crashes, among other hazards. The sustainability and capacity of the ultimate heat sink is also a relevant safety factor. The country experts are the ones who know best the characteristics of any national site.

Plant siting is followed by the design and construction licence; this requires a request from the future licensee and an evaluation by the regulatory body. Again there are guiding documents for both functions within the IAEA Safety Standard Series (IAEA, 2000a). Many countries now with operating nuclear power plants used the so-called 'reference plant concept' for licensing their first imported units. Under this concept, an imported plant had to have the same safety features as a defined plant in the country of origin of the project that has been previously licensed by the regulatory body of the exporting country. This process required the definition of a reference plant in a site with similar characteristics. This methodology could be used today, although it might be expected that the so-called certified designs will be built in new entrant countries.

Plant commissioning, operation and dismantling are also part of the licensing process (IAEA, 2000b). Again there are guiding documents in the IAEA Safety Standards Series. Commissioning is described in Chapter 22 of this book, operation in Chapter 23 and decommissioning in Chapter 24. Moreover, by this time the regulatory body in any entrant country will probably have increased its competence and human resources, and would also likely be part of the global safety regimen.

1.3.4 The need to justify nuclear power

Recognizing that nuclear power renders benefits but also creates radiation risks, Principle 4 of the IAEA Fundamental Safety Principles (IAEA, 2006) addresses the need for justifying the development of nuclear power and related activities by proving that such installations and activities 'yield an overall benefit'. This requires that all benefits coming from the operating plant and all significant consequences of such operation must be properly appraised. This is not an easy evaluation, and it will be described in Chapter 8 and Appendix 1 of the book. Chapter 16 develops in particular the social impacts of nuclear energy at the national and local levels.

The most valuable example of the justification of nuclear power has been provided by the UK Nuclear Industry Association (NIA) in its justification request for the four nuclear power designs which will possibly be built in that country: the Areva NP EPR, the Westinghouse Electric Company AP-1000, the General Electric/Hitachi ESBWR and the Atomic Energy of Canada Ltd Advanced CANDU ACR-1000. This request has been formulated in accordance with the justification regulations in the UK, the only country so far that has developed such regulations (NIA, 2008). At the time of writing the EPR and the AP-1000 designs have been justified, meaning that such designs could be built in any licensed site without the need for further stakeholder intervention at the national level.

Although the concept is normally applied in the use of radiation in medical practices, the application to nuclear power plants and related activities in an absolute way has been rendered difficult, although possible, as the UK example proves. Extensive literature has been published on comparing the relative benefits and risks between the different sources of electricity.

NEA has analysed the risks and benefits of nuclear power within the concept of sustainable development (NEA, 2007). There are also regulations in many countries requiring the analysis of environmental impacts for large construction projects, including nuclear power plants. Although they are not called justification exercises, many of these studies also contemplate social and economic impacts in an absolute way and by comparison with other solutions. All these efforts serve to develop tools and procedures for the justification report. This subject will be discussed in Chapter 8 of this book.

1.4 Conclusion

The growing optimism concerning nuclear energy which started to increase slowly with the twenty-first century has prompted many new countries to consider nuclear energy for their growing electricity demand; similarly countries with operating nuclear plants that had long postponed the construction of new plants became interested in building new units. Electrical companies, potential reactor suppliers, nuclear equipment manufacturers and architectural and engineering companies started to develop their capacities for a rather large new development of nuclear energy. International organizations provided positive outlooks into the future development of nuclear power, and high- and vocational-level educational institutions started to offer new nuclear education curricula. A sense of nuclear renaissance was growing around the world.

To cope with this new situation, in 2007 the IAEA developed considerations to launch a nuclear power programme, well considered by the Board of Governors, which initiated a large programme of assistance to interested Member States, the publication of documents on the needed national infrastructure for nuclear power, the celebration of workshops and international conferences and the development of large education programmes and teaching materials. At the same time, the IAEA continued the development of its Safety Standards Series and produced a consolidated set of Fundamental Safety Principles. INSAG also considered the safety needs for a nuclear renaissance which were reflected in its recent publications.

It was within this background that the need for this book was conceived. It was felt necessary to inform decision makers about the economic, industrial and educational infrastructure which is needed to support and maintain a reliable and safe nuclear industry; to transmit to governmental and private institutions and their employees the legal and technical requirements for a healthy and well-laid-out nuclear programme; and to inform society and members of the public that nuclear power is being developed within a worldwide net of administrative and physical barriers which prevent accidents with radiological effects and mitigate their consequences in case they occur, and that the operation of nuclear power plants does not cause unacceptable impacts on the health and safety of people and the environment.

To accomplish these desires, it was found convenient to base the book on the justification of nuclear power, as defined in Principle 4 of the IAEA Fundamental Safety Principles. This principle requires that, to be acceptable, nuclear power produces more benefits than detriments. In this first chapter it is concluded that the basic infrastructure needed to justify nuclear development and the elements to be used in any justification exercise are known and available. Details are found in the other chapters.

1.5 References

ACAD (2002), *Guidelines for the Conduct of Training and Qualification Activities,* ACAD 02–004, National Academy for Nuclear Training, Atlanta, GA.

IAB (2010), *First Semi-annual Report*, UAE International Advisory Board, Media Centre, Abu Dhabi (www.uaeiab.ae).

IAEA (1972), *The Structure and Contents of Agreements between the Agency and States Required in Connection with the Treaty of Non-Proliferation of Nuclear Weapons,* INFCIRC/153 (Corrected), IAEA, Vienna

IAEA (1994), *Convention on Nuclear Safety,* Legal Series No. 16, IAEA, Vienna.

IAEA (1997), *Model Protocol Additional to Agreement(s) between State(s) and the International Atomic Energy for the Application of Safeguard,* INFCIRC/540 (Corrected), IAEA, Vienna.

IAEA (2000a), *Safety of Nuclear Power Plants: Design,* Safety Requirements, IAEA Safety Standards Series No. NS-R-1, IAEA, Vienna.

IAEA (2000b), *Safety of Nuclear Power Plants: Operation,* Safety Requirements, IAEA Safety Standards Series No. NS-R-2, IAEA, Vienna.

IAEA (2002), *Preparedness and Response for a Nuclear or Radiological Emergency,* Requirements, IAEA Safety Standards Series No. GS-R-2, IAEA, Vienna.

IAEA (2003), *Site Evaluation for Nuclear Installations,* Safety Requirements, IAEA Safety Standards Series No. NS-R-3, IAEA, Vienna.

IAEA (2006), *Fundamental Safety Principles,* IAEA Safety Standards Series No. SF-1, Safety Fundamentals, IAEA, Vienna.

IAEA (2007), *Asian Network for Education in Nuclear Technology, ANENT,* Managing Nuclear Knowledge, IAEA Publications, Vienna.

IAEA (2008a), *Nuclear Security Culture, Implementing Guide*, IAEA Nuclear Security Series No. 7, IAEA, Vienna.

IAEA (2008b), *Preventive and Protective Measures against Insider Threads, Implementing Guide.* IAEA Nuclear Security Series No. 8, IAEA, Vienna.

IAEA (2008c), *Security in the Transport of Radioactive Material, Implementing Guide*, IAEA Nuclear Security Series No. 9, IAEA, Vienna.

IAEA (2009), *Nuclear Energy in the 21st Century. Addressing Energy Needs and Environmental Challenges*, Proceedings of a Ministerial Conference, Beijing, 20–22 April 2009, IAEA, Vienna.

IAEA (2010a), *International Status and Prospects of Nuclear Power,* Report by the Director General to the 2010 General Conference, GOV/INF2010/12-GC(54) INF/5, IAEA, Vienna.

IAEA (2010b), *Governmental, Legal and Regulatory Framework for Safety*, General Safety Requirements, IAEA Safety Standards Series No. GRS Part 1, IAEA, Vienna.

ICRP (1990), *1990 Recommendations of the International Commission on Radiological Protection,* ICRP Publication 60, Pergamon Press, Oxford, New York, Frankfurt, Seoul, Sydney, Tokyo.

INSAG (1992), *Probabilistic Safety Assessment,* IAEA Safety Series No. 75-INSAG-6, IAEA, Vienna.

INSAG (2003), *Maintaining Knowledge, Training and Infrastructure for Research and Development in Nuclear Safety,* INSAG-16, IAEA, Vienna.

INSAG (2006a), *Stakeholder Involvement in Nuclear Issues,* INSAG-20, IAEA, Vienna.

INSAG (2006b), *Strengthening the Global Nuclear Safety Regime,* INSAG-21, IAEA, Vienna.

INSAG (2008a), *Nuclear Safety Infrastructure for a National Nuclear Power Programme Supported by the IAEA Fundamental Safety Principles,* INSAG-22, IAEA, Vienna.

INSAG (2008b), *Improving the International System for Operating Experience Feedback,* INSAG-23, IAEA, Vienna.

NEA (2000), *Nuclear Education and Training: Cause for Concern,* NEA/OECD Publications, ISBN92-64-18521-6, Paris.

NEA (2004a), *Protocol to Amend the Convention on Third Party Liability in the Field of Nuclear Energy of 29th January 1960, as Amended by the Additional Protocol of 28th January 1964 and by the Protocol of 16th November 1982*, Atomic Energy Agency, Pairs.

NEA (2004b), *Nuclear Competence Building,* NEA No. 5588, NEA/OECD Publications, Paris.

NEA (2006), *Forty Years of Uranium Resources, Production and Demand in Perspective: 'The Red Book Perspective'*, Nuclear Energy Agency, OECD Publications, Paris.

NEA (2007), *Risks and Benefits of Nuclear Energy*, NEA No. 6242, Nuclear Energy Agency, Paris.

NEA (2008), *Nuclear Energy Outlook 2008*, NEA No. 6348, OECD Publications, Paris.

NEA (2010), *Uranium 2009. Resources, Production and Demand,* OECD IAEA, OECD Publications, Paris.

NIA (2008), *Justification Application: New Nuclear Power Stations*, Nuclear Industry Association, London.

NRC (1975), *Reactor Safety Study. An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants,* WASH-1400, NUREG-75/014, United States Nuclear Regulatory Commission, Washington DC.

OECD (2011), *Projected Costs for Generating Electricity in 2005–2010*, Nuclear Energy Agency, International Energy Agency, OECD Publications, Paris.

Tipping PH G (2010), Introduction to nuclear energy, and materials and operational aspects of nuclear power plants, chapter 1, page 5 in *Understanding and Mitigating Ageing in Nuclear Power Plants*, edited by Philip G. Tipping, Woodhead Publishing, Oxford, Cambridge, Philadelphia, New Delhi.

Index