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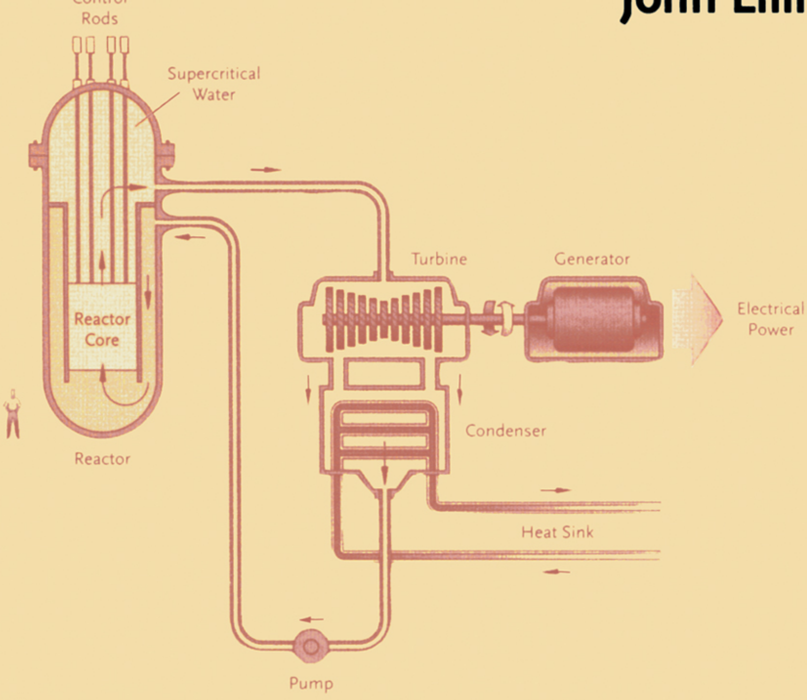
John Lillington



The Future of Nuclear Power

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Serco Assurance, Winfrith Technology Centre,
Dorchester, Dorset, UK

2004

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First edition 2004

Library of Congress Cataloging in Publication Data
A catalog record is available from the Library of Congress.

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

ISBN: 0-08-044489-X

♾ The paper used in this publication meets the requirements of ANSI/NISO Z39.48-1992 (Permanence of Paper).
Printed in The Netherlands.

Preface

During the last century, nuclear power has been established as a reliable source of energy in the major industrialised countries. It has a potentially important role in the future since it does not contribute to the production of 'Greenhouse' gases; a growing concern of continued fossil fuel power generation. The time is now appropriate to review the issues surrounding the future operation of current generation nuclear reactors and consider the potential offered by the new advanced reactor designs that have been proposed for the new century. The main purpose of the book is to present in a single volume the main issues of future civil nuclear power plant operation including the justification and incentives for future continuation, safety considerations and existing national strategies. The survey covers the entire major designs and their associated research programmes.

The evolution of the civil nuclear energy programme has seen the development of different generations of nuclear plant. In the US, the different generations have been designated as follows:

I - early prototype reactors in the 1950s & 1960s;
II - commercial power reactors in the 1970s & 1980s;
III - advanced light water reactor designs developed and certified in the 1990s, and;
IV - future generation nuclear energy systems.

Although this terminology has been introduced mainly in the context of US designs, it will be used more generally in this book in referring to the different generations of reactor systems in question.

The first part of the book reviews the commercial plants currently in operation (Generation II) and focuses on the issues concerning the future operation of these plants.

In the main, nuclear power plants have operated very successfully since the 1950s. Water reactors are the predominant type in the world today, mainly pressurised but there is also a significant fraction of boiling and heavy water reactors. The UK is an exception, where gas reactors are predominant. A brief survey of present day reactors is given in the first chapter.

There are wide ranging issues associated with the future of nuclear power. There are also very different perceptions of the benefits compared with the risks. There have been only a very small number of significant accidents, e.g. Three Mile Island and Chernobyl but these have had a major impact in limiting the expansion of nuclear power. The safety of plants for all aspects of operation, including the management of waste, is a public concern that needs to be addressed.

By far the most important pre-requisite for the continued operation of nuclear power plants is that they should remain safe and reliable. Improved safety has resulted from extensive evaluations of the few accidents that have happened together with a general improvement in all aspects of plant management.

Operational efficiency and reliable performance must be achieved to ensure competitiveness in the world market. Operating margins are being optimised, subject to safety limits, to enable maximum power output. Outage times for maintenance and refuelling are being minimised to produce high load factors.

The drive for improved safety and reliability is leading to improved maintenance operations and better monitoring techniques. The contribution of reactor diagnostics, through noise analysis, to both the safety and performance of operating reactors is increasingly recognised. The development of this technology for application to future and/or advanced plants is considered later in the chapters on experimental and theoretical research.

Modernisation programmes are in progress to improve the safety and performance of the older plants in operation. Some of these activities are being carried out in support of life extension, if there is an economic incentive to extend the life of current plants subject to meeting safety constraints. Many of the older VVER reactors are also being modernised. These include replacement of components to improve station performance, but also the back fitting of safety systems, in some cases to extend the design basis accident envelope.

Improvement of the fuel cycle is an important area of current attention. Holistic approaches are being considered to reduce costs over the whole fuel cycle. There is an increasing trend towards the use of high burn-up fuel. Mixed Oxide (MOX) fuels are also loaded into some present day plant. More is likely to be loaded into future reactors, beneficial as a means of reducing plutonium stocks. Advanced fuel cycles based on a thorium cycle could also be considered in place or uranium and plutonium cycles.

Technical solutions have been put forward for the management of waste and spent fuel. The high level waste component is largely contained in temporary on-site storage and some action will need to be taken to ensure continued safe containment. Further, there is a significant number of plants reaching the end of life over the next decade. This will increase the volume of decommissioning activity and the volume of material waste that will need to be managed.

Advanced reactor issues are considered in the second half of the book.

Design objectives are discussed for advanced reactors. There has been a range of different approaches adopted in the development of new advanced reactor designs to simplify the design and hence reduce cost. Evolutionary designs are being proposed, which represent relatively small perturbations from current technology. Other more innovative or revolutionary designs are also being considered that are substantially different from existing technology and these require major development investment.

Advanced reactors will need to meet continued demands for increased safety. Regulatory issues in regard to the potential licensing of advanced plant will be covered in the book. There are likely to be moves towards more harmonised approaches in licensing, perhaps enabled by an increasing tendency from vendors to seek design certification, as has been the trend recently in the US.

There are significant differences in the rates of nuclear power expansion and contraction across the different 'nuclear' countries across the world. A snapshot will be provided of the current status of nuclear industry in these individual countries in respect to their position on potential 'new build' or otherwise, likely preferred reactor systems, regulatory and political climate etc.

By taking account of operating experience and safety evaluations of current generation reactors, new advanced reactor designs have been proposed which are competitive and more economic than existing designs, incorporating standardised and simpler components. Increased reliability of safety systems will be a further requirement; an objective in many designs is to include a high degree of inherent safety, by taking advantage of the natural forces, e.g. gravity and natural circulation.

Evolutionary water reactor designs (Generation III) are being designed against these objectives and are most likely to be chosen for any 'new build' initiatives at least in the short term (e.g. 2005–2015). The book summarises the most likely candidates in a separate chapter.

A major feature of many evolutionary water reactors is a much greater adherence to inherently (passive) safe design principles. Because of their importance in some water reactors, these principles are discussed in a separate chapter. Some of these principles are also a characteristic of more revolutionary water reactor designs. Inherently safe principles are also adopted in some other (non-water) reactors.

Future generation reactors are covered towards the end of the book. These include both medium (e.g. 2015–2025) and long term (2025 onwards) deployment options. The medium term options include some evolutionary designs from reactor systems that have already been prototyped e.g. high temperature reactors). The long term options encompass the Generation IV systems referred to above. A review is provided on these advanced designs including super-critical water reactors, high temperature thermal and gas cooled fast reactors, liquid metal cooled fast reactors (sodium and lead) and molten salt reactors. These reactor systems collectively provide a capability for a wide range of applications, including electricity generation, plutonium and actinide management, heat applications and hydrogen production.

A discussion of Accelerator Driven Systems (ADS) is included in the book. These provide an alternative to the future generation critical reactors described above since they can be used in similar applications. These utilise spallation neutrons, generated from a proton beam incident on a target, in conjunction with a sub-critical reactor. Designs are being considered for electricity applications and particularly for the incineration of plutonium and the transmutation of waste.

Nuclear heat applications reactors, other than for power generation, are also briefly reviewed in a separate chapter. Non-power producing reactors for low temperature applications such as district heating and desalination are already in operation. High temperature applications for hydrogen production and for the chemical and process plant industries are not yet developed commercially but are seen as potentially important in the future. The future generation reactors, referred to above, would be candidates for these applications.

Several chapters towards the end of the book describe the extensive research programmes (experimental and theoretical) that are currently in progress for the purposes of ensuring the safety and reliable operation of current plant and design certification and safety assessment of advanced plant. These include reference to the available published material from reactor vendors and utilities and the more widely available research published by research institutes. Much of present research focuses on present day plant but much is also relevant to the needs of new reactor design developments.

Thus the book considers the significant designs over the range of different advanced evolutionary reactors through to the more exotic reactor designs being proposed, including fluidised bed and burn-up wave type reactors.

The book concludes with a discussion of likely longer term future requirements of a more general nature. This includes such topics as anticipated future energy and electricity requirements. It describes how new nuclear power producing plant could meet the requirements. The book finishes with a brief summary of non-nuclear power options in relation to the projection of possible overall nuclear development strategies in the next few decades.

J.N. Lillington

Acknowledgements

The author would like to acknowledge the work of many scientists and engineers that has been reviewed in the preparation of this book. The intention has been to reference all the individual contributions; apologies are given for any significant omissions.

I would like to express my gratitude to a number of colleagues within Serco Assurance, for many interactions and discussions on current issues relating to the future of nuclear power. These have been very beneficial to me in selecting the material for this book.

Appreciation is also expressed to Dr Salma Azmeh of Elsevier Ltd, Oxford for her advice and encouragement in the preparation of this work.

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Chapter 1

Present Generation Reactors

1.1. INTRODUCTION/OBJECTIVES

In the early days of nuclear power development, many different reactor types were considered and indeed prototypes were built. These included light water, heavy water, gas reactor and liquid metal-cooled fast reactor systems. The majority of the reactors in operation in the world today are light water reactors (LWRs) but there is also a sizeable fraction of heavy water reactors. Most of these reactors were built in the 1970s and 80s; only a few new reactors have been built during the last decade. Regarding other types, gas reactors continue to operate in the UK. There are only one or two prototype fast reactors still in operation, although interestingly both gas and fast systems are now starting to be reconsidered for next generation plants. These will be described in the succeeding chapters. Details will be given on the latest designs that are being proposed.

This is an introductory chapter to summarise briefly and review the designs of currently operating reactor systems. It presents the achievements of the technologies to date. It covers the principal reactors in operation today including light, heavy water, gas and other reactor types that have operated successfully, e.g. liquid metal-cooled reactors. The chapter defines the starting point for discussion of future designs in subsequent chapters. Thus, only the main features of the various reactor designs are highlighted below. Detailed descriptions of these reactors are included in a number of sources, see Leclercq (1986), Ramsey and Modarres (1998), Hewitt and Collier (2000) and Mounfield (1991).

The scale of current nuclear power plant operation worldwide is given in Table 1.1, which shows International Atomic Energy Agency (IAEA) data for 2002. This indicates that in 2002, there were a total of 441 units in operation in 30 countries, generating 358,661 MWe (Net) of electricity.

1.2. LIGHT WATER REACTORS

1.2.1 Pressurised Water Reactors

The pressurised water reactor (PWR) owes its origin to nuclear submarine reactor technology. The first civil PWR was built at Shippingport in the US and it entered commercial operation in 1957. This was a 60 MW (Net) reactor utilising high enrichment uranium fuel. This was soon followed by the Yankee Rowe plant, which included uranium oxide fuel and then other plants commenced operation both in the US and in Europe. Subsequent plants were progressively increased in capacity, in respect of the size of

Table 1.1. Nuclear power plant operation

Country	Nuclear units (number)	Total net electrical capacity (MWe)	Nuclear share 2002 (%)
Argentina	2	935	7
Armenia	1	376	41
Belgium	7	5760	57
Brazil	2	1901	4
Bulgaria	4	2722	47
Canada	14	10,018	12
China	7	5318	1
Czech Republic	6	3468	25
Finland	4	2656	30
France	59	63,073	78
Germany	19	21,283	30
Hungary	4	1755	36
India	14	2503	4
Japan	54	44,287	34
South Korea	18	14,890	39
Lithuania	2	2370	80
Mexico	2	1360	4
Netherlands	1	450	4
Pakistan	2	425	3
Romania	1	655	10
Russia	30	20,793	16
Slovakia	6	2408	65
Slovenia	1	676	41
South Africa	2	1800	6
Spain	9	7574	26
Sweden	11	9432	46
Switzerland	5	3200	40
Taiwan	6	4884	21
Ukraine	13	11,207	46
UK	31	12,252	22
US	104	98,230	20
Totals	441	358,661	

Data from Nuclear Technology Review (2003).

the components, the number of coolant loops (increasing from 1 to 4) and overall improvements in design. Large modern PWRs now generate typically up to 1300 MW (Net).

The basic components common to all PWRs are a reactor pressure vessel containing the core and the core barrel, primary circuit loops to convey the heat to steam generators, secondary loops to take steam to the turbine, together with a variety of other systems, e.g. control and safety systems. The primary side pressure is controlled by a pressurizer on one of the primary loops. The primary circuit is enclosed in a containment. There have been various differences in the design of these major components across the various vendors but the fundamental principles are common. Figure 1.1 shows a schematic of the modern Sizewell B PWR.

Principal PWR vendors included Westinghouse, Babcock and Willcox, Combustion Engineering in the US; in Europe, Framatome in France and Kraftwerk Union (KWU) in Germany.

Modern PWR cores comprise assemblies containing fuel rods and absorber rods in a vertical bundle. The rods are arranged in a lattice of 17×17 positions. Of these, about 264 positions are occupied by Zircaloy-4 clad fuel rods of about 3% enriched U-235, the remainder of positions are occupied by absorber rods.

The vessel contains light water at a sufficiently high pressure to prevent boiling. The discharge temperature and pressure are about 320°C and 15.7 MPa, respectively. Reactivity is controlled by positioning of the control rods and by managing an appropriate concentration of boron in the coolant. Water is pumped to the steam generators, from which heat is transferred to the secondary side operating at a pressure in the region of 6–8 MPa. Steam produced is passed through moisture separators and dryers before entering the turbine generator. It is subsequently condensed, reheated and returned to the steam generators and the cycle is repeated. There are some differences in detail between different designs.

Typical features of some of the principal designs are as follows. In the Westinghouse PWR for example, the steam generators consist of inverted U tubes immersed in water within the secondary side loop. Other designs, e.g. Babcock and Willcox incorporate once through steam generators, which enable the steam to be slightly superheated.

Other designs exhibit different distinctive features, e.g. in the KWU reactor there are no penetrations in the lower head of the reactor vessel. The KWU design also incorporates a

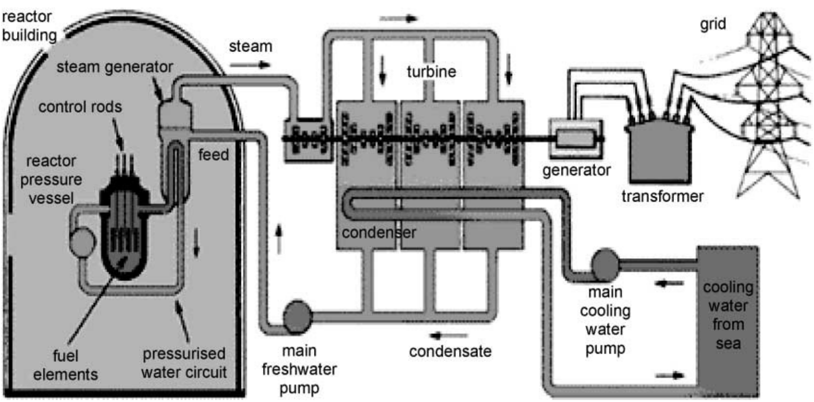

Figure 1.1. Sizewell B pressurised water reactor. Source: http://www.british-energy.co.uk.

spherical (as opposed to a cylindrical) containment principle. It includes a steel containment structure encompassing the primary system, which is itself enclosed in a reinforced concrete building.

Framatome have introduced boron carbide control rods in contrast to the silver-indium-cadmium rods of other designs to enable greater flexibility of control. The company has also pioneered further improvements in respect of extended fuel cycles and the use of MOX fuel.

PWRs have operated very successfully over many years. A wealth of experience has therefore built up that has resulted in improved operational, cost effectiveness and safety. PWRs are the most widely used plants in operation in the world today, both in terms of the number of units, the quantity of electricity produced and in their distribution worldwide. Table 1.2 indicates that by the end of the 1990s, PWRs dominated the generating capacity of nuclear reactors worldwide; there are about 204 units producing a gross capacity of 203,228 MWe in 15 different countries. This trend continues today.

PWRs are refuelled off-load. During refuelling, a third of the spent fuel is removed, the remaining two-thirds is relocated to different parts of the core and new fuel is loaded. The core is arranged to provide optimal performance. A disadvantage of the PWR is that it can only be fuelled off-load, which means that the reactor has to be down for 4–6 weeks. During the outage, maintenance operations can be carried out. Typically, once every 3 years, the pressure vessel and internals are inspected, which means that all the fuel has to be removed and this outage might take up to 3 months.

In terms of running costs, these reactors along with most other current plants require some degree of uranium enrichment, and therefore fuel costs are relatively high. Against this they utilise abundantly available water both as moderator and coolant – the cost of these being low. Overall PWRs can compete economically with fossil fuel plants over many years.

Table 1.2. Current generation reactors

Reactor type	Units in operation (number)	Countries of operation (number)	Gross electrical capacity (MWe)
PWR	204	15	203,228
BWR	95	11	82,920
VVER	47	8	31,852
RBMK	14	3	14,600
PHWR	34	6	19,555
Magnox	21	2	3952
AGR	14	1	9164
FBR	7	5	2547
Other	12	3	590

Data from 1997 World Nuclear Industry Handbook (1997).

PWRs have a relatively complex technology requiring diverse safety systems to guard against major loss of coolant accidents. Modern PWR designers have recognised this weakness and have attempted to simplify the complexity (and hence reduce capital costs) in new proposed designs. These are discussed in later chapters in the book.

1.2.2 Boiling Water Reactors

Boiling water reactors (BWRs) were first developed in the US by the General Electric Company. The first commercial BWR, Dresden, sold to the Commonwealth Edison Company, was a 200 MW plant commissioned in 1960. This was followed by subsequent orders in the US, Europe and Japan. Ratings were increased up to the 1300 MW plants in operation today. Other vendors developed designs, independently of the US, notably Asea-Atom, later ABB Atom, in Sweden, Figure 1.2.

The characteristic feature of BWRs compared with PWRs is that boiling occurs within the core. Due to the axially changing void fraction, the axial flux becomes asymmetric. After drying in moisture separators (as in a PWR), the steam is passed directly to the turbine. The loop is completed by condensing the steam; the condensate is then returned to the reactor vessel. The Forsmark 3 BWR loop is shown in Figure 1.2.

BWRs burn uranium oxide fuel at a typical enrichment of around 2%. Fuel rods are grouped in a square lattice of 6×6 up to $8 \times$ rods, the full assembly being smaller than in the PWR. The enrichment within the rods depends on their position in the fuel assembly, the reason for this being to correct for the effects of water spaces between the fuel assemblies. Reactor control is achieved with control rods inserted from the bottom of the core. The absorber material in the rods is boron carbide.

There have been changes in the main recirculation systems employed in BWRs during their evolution. For example, in some of the older BWR designs, the water is circulated by external pumps, one pump on each loop external to the vessel. In the more recent designs, the tendency is to utilise internal pumps, to avoid the risk of loss of coolant in the event of an external line break. General Electric employed an intermediate system with both external and internal pumps. Reactor power can be controlled by altering the flowrate

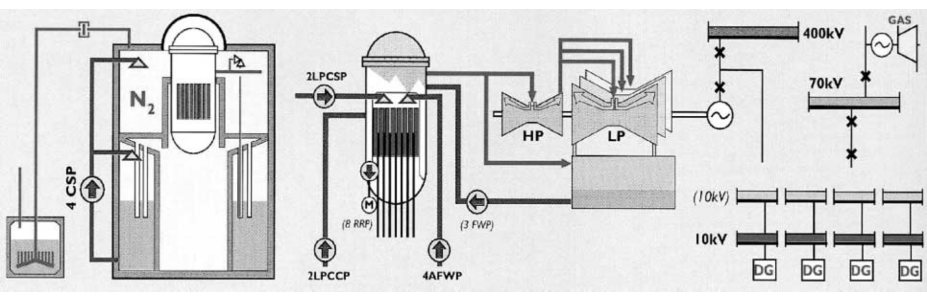

Figure 1.2. Forsmark 3 boiling water reactor. Source: http://www.okg.se.

since this affects the core water temperature and steam bubble level, thereby affecting the neutron moderation.

BWRs operate at a lower pressure than PWRs, typically 7–8 MPa. BWR vessels are generally larger than PWRs, which is a disadvantage, despite having the advantage of a single cycle system. The turbine area in a BWR has to be monitored to ensure that health physics regulations are satisfied. Radioactive products can be transported in the steam, from a failed fuel rod for example.

BWRs are constructed with a leak tight containment, which is designed to withstand the load from a large break in the coolant or steam system. Safety systems have the provision to inject water directly into the reactor vessel to cool the fuel. Containment pressure increase is relieved via condensation in water filled areas. There is an additional cooling system to spray the chamber surrounding the reactor vessel.

The design of containment has evolved through the years, mainly in relation to the designs of the dry well that surrounds the reactor and the wet well that contains the water for pressure suppression in the event of a reactor vessel penetration failure. For example, General Electric developed the Mark I, II and III design containments, the principal driver being to simplify design and increase capacity. Six different models, BWR 1–6, have been developed, incorporating different pump configurations, increased fuel assembly arrays and power density.

BWRs exhibit many of the advantages and disadvantages associated with PWRs. They have also been operated very successfully over a long period of time and much experience in operation has been accumulated. They are fuelled off line, utilise similar fuel coolant and moderator and have relatively complex technology, albeit that the single cycle system of the BWR is clearly a simplification of the two loop cycle of the PWR (and hence capital costs tend to be somewhat lower). Comparative data for the PWR and BWR and also for the other reactor designs are given in Table 1.3.

1.2.3 VVER Systems

The first Soviet designed VVER reactor based on PWR technology was a 265 MW plant commissioned at Novovoronezh in 1964. The first generation of VVER reactors were of 440 MW capacity (VVER-440/230) and 10 such plants were built in Russia and Eastern Europe. These plants had relatively limited safety features and were followed by a series of 14 second-generation plants (VVER-440/213) with much improved safety features. Two 445 MW plants were also built at Loviisa in Finland in 1977. These included a strong steel-lined reinforced concrete containment, with ice compartments.

The 440 MW VVER reactors have many features in common with Western style PWRs but they also have some differences. For example, they include six coolant loops with horizontal steam generators that are of generally smaller capacity than Western designs. The core lattice is hexagonal, with typical enrichments of 2.2–3.6% uranium-235.

Table 1.3. Representative technical data

Plant	PWR (George and Board, 1987)	BWR (Handbok over processainband vid störningar I svenska kokarreaktorer, 1987)	VVER (IAEA, TC/RER/9/004, 1994)	RBMK (Alemenas *et al.*)	PHWR (1997 World Nuclear Industry Handbook, 1997)	Magnox (1997 World Nuclear Industry Handbook, 1997)	AGR (IAEA Publications)
Model	Westinghouse 4-loop	Internal pump	440/213	LWGR	CANDU	Gas	Gas
Reference plant	Sizewell B	Forsmark 3	Bohunice V2, 3 & 4	Ignalina 1 & 2	Darlington 1–4	Wylfa 1 & 2	Hartlepool 1 & 2
Nominal electrical output	1245	1190	440	1500	935	570	666
Coolant	Water	Water	Water	Water	Heavy water	Carbon dioxide	Carbon dioxide
Moderator	Water	Water	Water	Graphite	Heavy water	Graphite	Graphite
Fuel	Oxide	Oxide	Oxide	Oxide	Oxide	Metal	Oxide
Coolant pressure	15.8	7.0	12.3	7.0	10.6	2.8	4.2
Coolant outlet	325	286	297	284	313	370	675
Containment	Steel-lined pre-stressed concrete	Pressure suppression pre-stressed concrete with liner	Reinforced concrete with inner steel liner		Reinforced concrete		Reactor building

The system pressures are somewhat lower than for PWR ~ 12 MPa. Control rods are also hexagonal and replace a whole core assembly.

Third-generation VVERs of 1000 MW design are also operational, the first being built in 1981 in the Ukraine. There are around 18 plants in operation. These include the most advanced safety features among the VVER designs, including sealed containments. Figure 1.3 shows the Temelin 1 reactor in the Czech Republic.

The VVER-440/230 plants have a number of design limitations. In particular, there are limited emergency core cooling systems and large pipe guillotine breaks were not included in the design basis. These plants do not include a strong containment to enclose the reactor. There were limitations in the control instrumentation and in the design of control rooms with limited protection for operators in the event of a large release of radioactivity. These shortcomings have been recognised by the IAEA and other bodies, e.g. the Organisation for Economic Co-operation and Development (OECD) and the European Commission (EC), and improvements have been made. Nevertheless significant safety issues remain for these types of reactors (Lederman, 1995).

The VVER-440/213 plants are much improved in their design and safety concept. For example, their safety systems can mitigate the large guillotine break, they also have sealed containments (or confinements) to localise accidents via a suppression system. The VVER-440/213 plants have been in operation since the early 1980s. They have shown

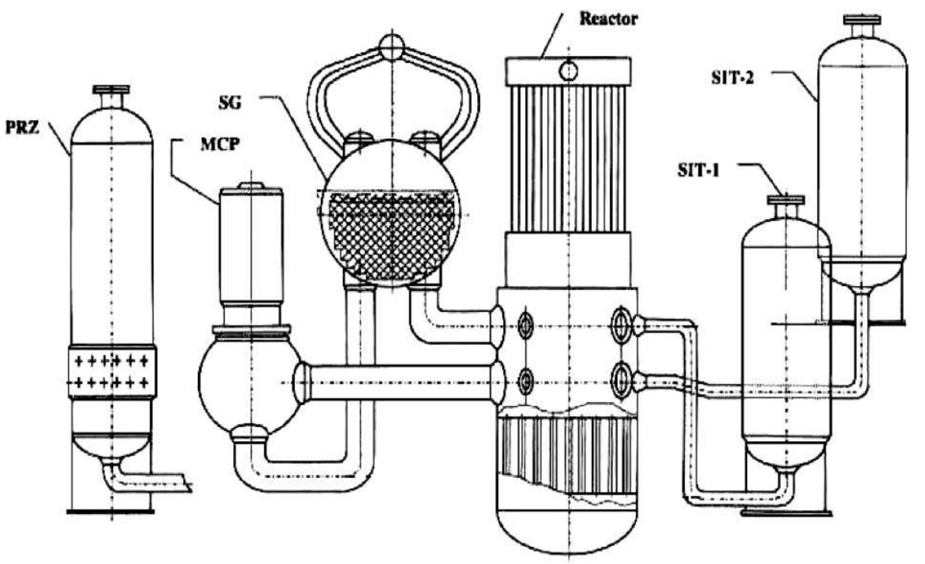

Figure 1.3. Temelin 1 VVER-1000. Source: EUR 20056 (2001).

good availability and have a good safety record (e.g. in terms of radiological safety and event frequency).

VVER 1000 plants are designed consistently with standard international practice. As for Western plants, they employ the well-established defence-in-depth concept and include redundancy, diversity, physical separation and fail-safe principles in design. However, the standards of manufacture and construction of some units have been questioned. There have also been questions on the power stability, instrumentation and control room operation of these units.

As for other PWRs, VVERs are also refuelled off-line.

1.2.4 RBMK Systems

RBMK graphite moderated reactors were the first nuclear power plant designs developed in the former Soviet Union. The first plant was built at Obninsk in 1954 but the first units to provide a significant power capability were commissioned at Beloyarsk (Unit $1 \equiv 102$ MW in 1964 and Unit $2 \equiv 185$ MW in 1976). This led the way to the development of twin 1000 MW designs and two 1500 MW units (the latter at Ignalina in Lithuania), see Figure 1.4.

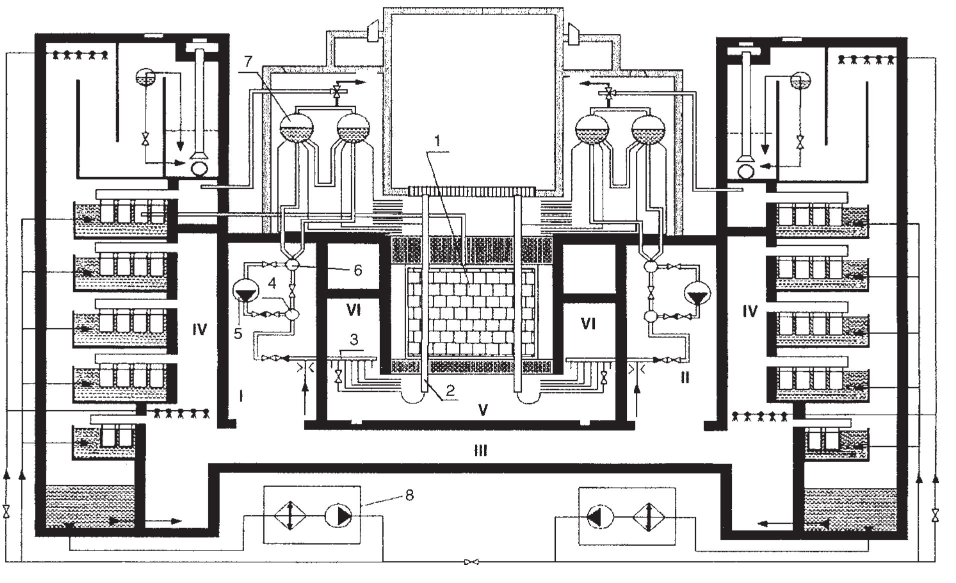

Figure 1.4. Ignalina 1 RBMK-1500. 1: graphite stack; 2: fuel channels; 3: group distribution header; 4: pressure header; 5: main circulation pump; 6: suction header; 7: drum separator; 8: condensation tray cooling system. Compartments: I, II: reinforced compartments (left- and right-hand sides) enclosing the major components of the main circulation circuit (main circulation pumps, suction headers, pressure headers and downcomers); III: reinforced steam removal corridor; IV: towers; V: under-reactor compartment; VI: compartments of the lower water piping. Source: Dundulis *et al.* (2003).

RBMK reactors employ a direct cycle boiling water pressure tube concept, which was favoured because it avoided the problem of fabricating large pressure vessels. The pressure tubes pass through the graphite core, which is about 12 m high comprising the graphite blocks. The blocks are penetrated by the Zircaloy alloy pressure tubes, each about 88 mm internal diameter and 4 mm thick.

In the 1000 MW design, there are 1663 channels, each containing two fuel assemblies, 3.64 m long. The fuel assemblies consist of 18 pin clusters. Each pin contains about 2% enriched uranium dioxide pellets in Zircaloy alloy cladding, 13.6 mm outside diameter and 0.825 mm thick.

The light water coolant is at about 7 MPa pressure. The inlet temperature of the water is 270°C, the quality of the existing steam water mixture is about 14%.

The coolant system consists of two identical loops. These loops feed into two steam drums. Each loop has four primary circulating pumps, of which one is usually for standby. The dry steam is passed to one of two 300 rpm 500 MWe turbine generators. After purification, the condensate is returned to the steam drums via electrically driven feed pumps.

About 5% of the heat is dissipated in the graphite. This is transferred to the fuel channels via graphite rings, which allow good heat transfer between the pressure tube and the graphite blocks. The graphite temperature should not exceed 700°C.

One important feature of the RBMK is that it has a positive void coefficient. Clearly, the net effect of the positive void coefficient and negative fuel temperature coefficient is an extremely important factor, which will depend on the power level. The RBMK therefore is a sensitive reactor to control. At full power, the negative fuel temperature coefficient dominates, but at low power this is not true.

Channels for the control and shutdown rods also pass through the graphite blocks. For the reactor control and protection there are 211 solid absorber rods that are divided into rods with different operations of control.

The RBMK is refuelled at full load.

The primary circuit is contained in a series of compartments that perform the function of a containment in the event of an accident. Each compartment has a design pressure of about 0.45 MPa.

As a consequence of the Chernobyl accident, a number of modifications have been added on other units. These include, improved rate of control rod insertion, automatic shutdown systems to prevent low-power operation and also the problem of positive void coefficient has been mitigated by the fitting of fixed neutron absorbers together with increased fuel enrichment.

There are 15 RBMKs in operation but there is still international concern over the safety of these reactors. Nevertheless, these plants account for relatively high percentages of the total nuclear generating capacity in Russia and Lithuania (NB. in the Ukraine, all Chernobyl plants are now shut down).

1.3. HEAVY WATER REACTORS

Pressurised heavy water reactor (PHWR) concepts have been developed in a number of countries, including Canada, Japan, France, UK and others. Interest in heavy water as a moderator arose because it overcame the problem in LWRs of relatively high absorption of neutrons, enabling the reactor to operate at lower enrichment, or even with natural uranium.

However in the UK, the steam generating heavy water concept was not taken forward to commercial operation because the economies of scale were not favourable in comparison with alternatives.

1.3.1 CANDU Designs

The Canadian designed Canadian deuterium uranium (CANDU) reactors used natural uranium as a fuel, by employing heavy water as both moderator and coolant. The first CANDU, NDP2-Rolphton of just 23 MW, entered commercial operation in 1962 and a number of 2, 3 and 4 unit plants evolved of commercial power capacity, individual units delivering power in the range 500–800 MW. A schematic of the Darlington PHWR is shown in Figure 1.5.

CANDU reactors consist of horizontal pressure tubes constructed with Zircaloy alloy. They pass through a large vessel (Calandria) filled with heavy water (deuterium oxide) at low pressure and temperature. Uranium oxide pellets are sealed in Zircaloy alloy cans, which are assembled in bundles or fuel assemblies. In a 500 MW plant, each bundle has

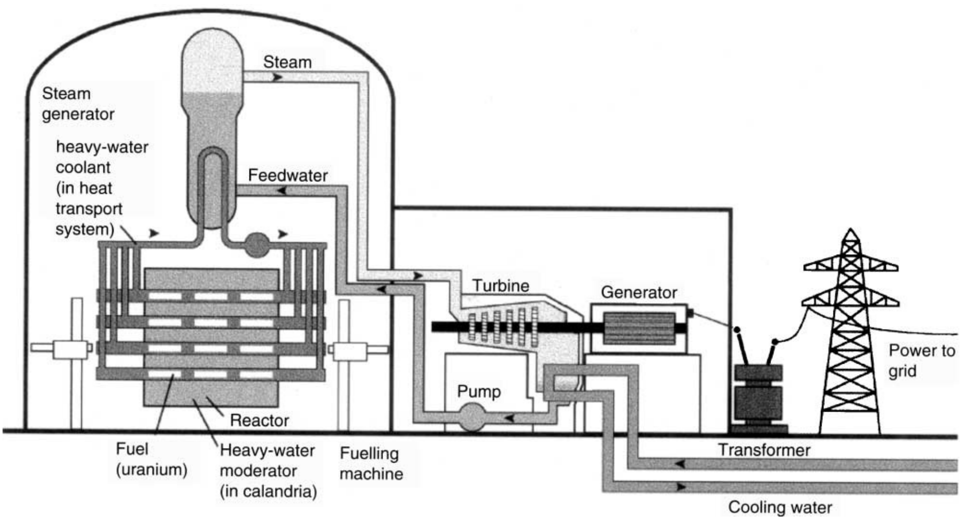

Figure 1.5. Darlington PHWR. Source: Hedges (2003).

about 28 elements. There are about 4860 bundles in total with 12 or 13 such bundles in each pressure tube.

The heat generated is removed by heavy water at about 9 MPa, a sufficiently high pressure to prevent boiling. The water circulates around the fuel elements and passes to a steam generator, a similar principle to the PWR and BWR concepts.

The CANDU reactor is controlled by cadmium absorber rods. When fully inserted, these also provide the shutdown margin. In addition, the reactor can be shut down by voiding the cold heavy water from the reactor core. Vertical steel 'adjusting' rods are used to smooth out the sometimes uneven power distribution due to the use of different burn-up fuel segments, even within one fuel channel.

As for the PWR, the CANDU primary system is located in a concrete containment building, of sufficient strength to accommodate a large coolant system break within its design basis. Modern CANDUs are connected by valves to a large vacuum building. In the event of an accident these enable steam to pass from the affected containment to the vacuum building.

The CANDU reactor has a low volumetric power density, about 10 times lower than a PWR, despite fuel ratings that are comparable with PWR. It also has the lowest fuel costs because of utilisation of natural uranium. Against these cost benefits, the CANDU reactor needs considerable quantities of heavy water.

The CANDU reactor does have a number of advantages. It has on-load refuelling and hence has very high load factors. The plant has high availability and high reliability also. Since the design incorporates individual tubes, there is no requirement for a large pressure vessel. From a safety perspective, the reactivity excess is smaller than in reactors employing enriched fuel and hence power excursion transients are less likely.

In terms of disadvantages, the CANDU has a very large core (compared with a PWR or BWR) to achieve a similar power output.

1.4. GAS-COOLED REACTORS

1.4.1 Magnox Reactors

The first Magnox reactors built in the UK were at Calder Hall and Chapelcross. These were just 50 MW plants; eight units being built in total which were commissioned between 1956 and 1960. These first plants were originally envisaged for the purpose of producing plutonium but were also operated to produce electricity. They were followed by a series of higher rated plants commissioned between 1962 and 1971. The most highly rated plant was Wylfa at 590 MW operating at a gas pressure of about 27 bars. Many of the earlier plants are now shut down but the later plants are still in operation.

The Magnox reactor core consists of a 'pile' of graphite blocks or bricks which contain channels. Carbon dioxide at a pressure of typically a few tens of bars flows through these channels, which also contain the fuel elements or control rods. The fuel elements consist of natural uranium bars clad with a magnesium alloy known as Magnox. These are machined into a 'herringbone' pattern in order to optimise heat transfer. A metallic fuel was adopted; i.e. natural uranium was used. The magnesium alloy was specifically chosen because it did not have a significant absorption of neutrons, enabling natural, rather than enriched uranium to be used.

Typical geometric and operating parameters are defined to limit the internal temperature of the elements to about 650°C, a critical temperature at which deformation of the uranium crystal lattice occurs. Similarly, the can temperature is limited to 420°C, associated with the use of Magnox alloy. A typical Magnox core is about 8 m high and 14 m in diameter. The core exit gas temperature is about 400°C.

On exiting the core, the coolant flows directly to the steam generator and then is pumped back to the reactor. The efficiency of the steam cycle is around 31%.

In the early Magnox designs, the vessel was made of steel and the steam generators (heat exchangers) were external to the pressure vessel. In Oldbury and Wylfa, the heat exchangers were placed inside the pressure vessel, constructed with pre-stressed concrete (Smitton, 2000).

Magnox reactors have in general operated very successfully in the UK over a period of many decades. However, from an economic perspective they have a low power density with high fuel costs.

1.4.2 Advanced Gas Reactors

Advanced gas reactors (AGRs) were designed to overcome some of the inherent limitations of the Magnox design. The main problem with the Magnox design was the low power density, pressure and operating temperatures.

The first prototype AGR was built at Windscale in 1962. The commercial AGRs that were subsequently built were twin 620–660 MW plants. Seven stations were built; these entered commercial operation in the late 1970s and 1980s. The first industrial plant was at Hinkley B commissioned in 1976. These plants ran into difficulties during their construction and design phases due to problems that were both industrial and technical. In all, three different industrial groups were commissioned with different design approaches. The Dungeness B loop is shown in Figure 1.6.

The AGR uses carbon dioxide as a coolant, like the Magnox plants, but in order to achieve higher coolant pressures ($\sim$40 bars) and temperatures (outlet temperatures $\sim$650°C), a new fuel design was required. The fuel became uranium dioxide pellets, inside stainless steel tubes.

AGR fuel had to be enriched to about 2.3% uranium-235 in order to overcome the significant neutron absorption of the stainless steel fuel cans. With this enrichment, it was

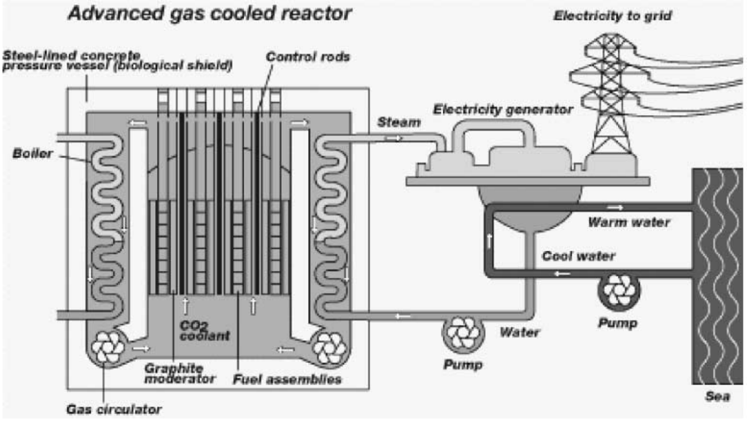

Figure 1.6. Dungeness B advanced gas reactor. Source: http://www.british-energy.co.uk.

possible to achieve a 3-fold increase in volumetric power density with an average fuel rating of 4-fold increase compared with the best Magnox stations.

The more onerous pressure and temperature operating conditions created difficulties for the designers associated with vibration, chemistry (corrosion) and concrete insulation problems.

In the AGR, the coolant gas is circulated from the core to steam generators. These are mounted inside the pre-stressed concrete pressure vessel. These steam generators comprised 4 or 8 steam raising units. Good efficiencies are achieved as high as 40%. The steam generators provide steam at around 170 bars and 560°C, conditions that are comparable with those in an efficient fossil fuel plant.

A problem of concern for the AGR designers was attack of the graphite moderator by the carbon dioxide gas, which could oxidise the graphite and reduce its strength. This was overcome via controlled coolant chemistry with an appropriate level of water vapour content together with a small concentration of methane. This was however a delicate balance, because too much methane could result in carbon deposition on the fuel elements and consequent degradation of heat transfer.

AGRs can be refuelled on load and the fuel can remain in the core for long periods, up to 5 years. They have high fuel efficiency, up to about 40%; they have a more efficient use of fuel compared with LWRs. The AGR has a number of inherent safety features; e.g. the graphite has a large thermal capacity in the event of a primary circuit rupture.

A disadvantage of AGRs has been the limited investment of international vendors to support their technology. This, coupled with the lack of standardisation, has led to higher capital costs. It has not competed successfully outside of the UK in comparison with the PWR and BWR.

1.4.3 High Temperature Reactors

There is clearly a strong incentive to maximise the thermodynamic efficiency of nuclear power plants and one way of achieving this is to increase the temperature of the coolant. From the early days of nuclear power there has been considerable interest in helium cooled high temperature reactors (HTRs).

A 20 MW prototype, the Dragon reactor, was built and operated at Winfrith between 1964 and 1975. The plant was operated as part of an international OECD co-operative programme. Although, there were plans for a follow-on programme to Dragon, these were not pursued.

Another 13 MW prototype, the AVR, was built in 1966 at Jülich in Germany based on the 'pebble bed' design. In this design, the fuel consists of particles of thorium or uranium dioxide fuel surrounded by carbon. These particles are a fraction of millimetre in diameter and are bonded into balls. Following AVR, a 295 MW plant was built at Schmehausen in Germany in the early 1980s and this achieved power in 1985.

In this design, the core is filled with approximately 675,000 spherical graphite fuel particles. The helium coolant is pressurised to about 40 atm and exits the core at 750°C. Heat is transferred to water and steam, circulating in stainless steel tubes within the helium. Steam passes to the steam generator at 530°C and 181 atm.

Another model, taken forward in the US was the prismatic core design. In this design, the fuel particles are formed into cylindrical rods and placed in hexagonal graphite blocks with coolant channels. An initial 40 MW prototype designed by the General Atomic Company was built at Peach Bottom in the US. This operated from 1966 to 1974. It was followed by a 330 MW prototype at Fort Saint Vrain, which came onto the grid in 1976. Here, 10,000 fuel particles are fixed in a graphite matrix with 210 fuel channels and 108 helium channels. The helium is at 48 atm, there are 1482 fuel blocks. Somewhat higher coolant outlet temperatures were achievable with this design.

The helium-cooled reactors have a number of attractions in principle. Helium is a preferred coolant to carbon dioxide in the presence of graphite since it is inert and therefore does not oxidise graphite – a problem at higher temperatures in carbon dioxide-cooled reactors.

Another attraction of the helium-cooled reactor designs discussed above was that they could be used to produce fissile material from less useful uranium fertile material. Uranium-238 is converted to plutonium-239 and uranium-233 to thorium-232. There was, therefore, the possibility of achieving very high burn-up with targets up to 100,000 MW days tonne^{-1}.

Difficulties were encountered in the early days of these reactors and new orders were not placed following these prototypes. However, there has been a recent revival of interest in HTRs in recent years, e.g. the ESKOM project in South Africa.

At the time of writing, there is no commercial power plant of this type in operation. However, this type of reactor is one of the designs under consideration in the US Generation IV programme. These designs are discussed in detail in subsequent chapters.

1.5. LIQUID METAL-COOLED REACTORS

1.5.1 Fast Reactors

The first such reactor to generate electricity was the US Experimental Breeder Reactor 1 (EBR 1). This started in 1951 with a capacity of 200 kWe. It was fuelled by highly enriched uranium-235. In common with future fast reactor designs, the core was small and compact. The fuel pins were just 1.25 cm in diameter. The core consisted of 217 pins in a hexagonal lattice. The coolant was a sodium/potassium alloy, surrounding the central region was a blanket region containing rods of natural uranium. EBR 1 operated until 1963 and yielded considerable information on liquid metal fast breeder reactor (LMFBR) technology. A second reactor EBR 2, 15.7 MW, was also built on the Arco site in Idaho.

A 60 MW commercial reactor, Enrico Fermi 1 went critical in 1963. This reactor underwent a serious loss of coolant accident in 1966. It restarted for a few years but was finally shut down in 1970.

The US fast reactor programme continued with various test facilities until 1983, e.g. the southwest experimental fast oxide reactor (SEFOR) at Arkansas, the transient reactor test experiment (TREAT) at Argonne and the fast flux test facility (FFTF) at Hanford.

Within Europe, the United Kingdom atomic energy authority (UKAEA) built several research reactors before the Dounreay fast reactor (DFR) was commissioned and became critical in 1959. DFR had a modest electrical capacity of 14 MWe. It was closed down in 1977. The prototype fast reactor (PFR) had an electrical output of 254 MWe and entered service in 1975. It operated for over a decade before being shut down.

This sodium-cooled fast reactor was a pool type design. A pool of sodium is contained in a vessel with sodium pumped through the core by pumps contained within the pool. The hot sodium then passes through an intermediate heat exchanger; transferring heat to a second sodium-cooled loop. The latter transfers heat to a water/steam loop via the steam generator. This tertiary loop system ensures that any radionuclides produced in the primary vessel remain in the vessel and are not transferred to the steam generator.

In this type of reactor design, the reactor functions on fast neutrons, there is no moderator.

In France, a similar 250 MW prototype was also built (Phénix), which was then followed by a commercial sized plant (Superphénix), the latter commissioned in 1986 (but now closed down permanently).

Other countries have explored the production of fast reactors, e.g. Germany, Japan, India and the former Soviet Union.

LMFBRs have a number of advantages. Liquid metals have desirable thermophysical properties. The coolant has a low melting point, coolants can be chosen, e.g. sodium and potassium, which have low neutron absorption. Sodium has a high thermal conductivity, albeit a lower specific heat than water and it has a high boiling point, etc.

LMFBRs also suffer from a number of disadvantages and problems. There are concerns over the use of sodium since it is highly reactive to oxygen and water. There is a potential problem of isolation of the sodium and water-cooling loops. There have been problems in the steam generators of fast reactors.

In recent years the development of fast reactors at the commercial scale has slowed down. Nevertheless, the potential for fast reactors exists and is still under review in some countries. Fast reactors are again under consideration in the US Generation IV programme.

Historically, the fast reactor has always been considered in relation to its fuel cycle, its ability to burn and breed plutonium. In addition, most reactors produce plutonium, in differing amounts, which can in principle be recovered for utilisation in a fast reactor fuel cycle. However, there are safety and economic issues associated with fuel reprocessing, these are considered later. Plutonium can also be burnt in thermal reactors to improve the economics of the thermal fuel cycle.

1.6. FUSION

One of the limitations of fission power is that it depends on uranium (and possibly thorium) reserves, which are a finite resource. The utilisation of fast reactors and accelerator-driven reactors, especially if used in a thorium fuel cycle (since the reserves of thorium are greater those of uranium) would substantially increase the energy available from this resource, but nonetheless the statement remains true at least in principle. The goal of generating almost limitless energy from the fusion of appropriate light isotopes of hydrogen or lithium has been a dream of scientists for many years. This dream is not yet realised but it is deemed that sufficient progress has been made towards achieving controlled fusion, that fusion reactors deserve a mention in this introductory chapter on present generation reactors.

A significant problem in the development of a fusion reactor has been the confinement of the nuclei in order that the fusion reactor can proceed in a controlled manner. Fusion with a positive energy balance is only possible at very high temperatures. These must be so high that the thermal agitation of the atoms is sufficiently energetic that the electrostatic repulsion of the positively charged nuclei can be overcome, enabling collisions to occur.

A number of different fusion reactions have been postulated between the isotopes of hydrogen, helium and lithium. However, the majority of research efforts have concentrated on the deuterium–tritium reaction. This is the easiest reaction to achieve. Nevertheless, temperatures must be of the order of 100 million degrees.

There is also a confinement criterion, which requires that the period of the confinement time and the neutron density must exceed a stringent limit (Lawson Criterion).

Focus has concentrated on essentially two types of confinement, magnetic and inertial. Of these, magnetic confinement has received the most attention.

In magnetic confinement, a strong external magnetic field consisting of a high density of field lines is imposed. In a toroidal system, the field is circular such that the nuclei in the deuterium–tritium mixture travel in helical paths around the magnetic lines of force. This gives rise to the shape of a torus. In an ‘open’ system, the field lines are not closed but a series of magnetic coils are arranged to reflect particles back into the centre of the field. These are referred to as ‘magnetic mirrors’. The challenge with either of these methods is that the plasma should not contact the confining vessel, otherwise the temperature will fall.

In inertial confinement, pellets are made from a mixture of deuterium and tritium in a mixture frozen at about 15 K. These are then irradiated either by very powerful laser beams or by electron (or ion) beams. These compress and heat the material to fusion level temperatures; inertia results in very high densities for very short periods of time (order of a nanosecond). However, there are practical difficulties with this approach associated with the laser efficiency and engineering problems in achieving a continuous power output.

With regard to on-going research, the Tokamak system has probably attracted the most attention. The ideas were originally conceived in the former Soviet Union. The system is based on the closed magnetic field configuration in the shape of a torus. The Joint European Torus (JET) project in the UK has made progress in generating significant amounts of power, in 1991, 2 MW were achieved. However, the break-even point, i.e. the generation of as much fusion power as is required in heating up the plasma has not yet been achieved. The trend is generally for larger Tokamaks in the quest to achieve higher and higher temperatures and conditions that will satisfy the Lawson criterion.

In the US, the Lawrence Livermore Laboratory and the Los Alamos Laboratory are carrying out work on inertial confinement. Lower fractions of energy produced against input have been produced in comparison with the Tokamak approach.

Significant progress in fusion technology has been achieved to date and these have been described in this chapter. For the next generation of Tokamaks, the resources of the interested nations are likely to be pooled in the International Tokamak Experimental Reactor (ITER) Project.

REFERENCES

Alemenas, K., Kaliatka, A. & Uspuras, E. (1997) *Ignalina RBMK-1400, A Source Book*.

Dundulis, G., Kaliatka, A. & Rimkevicius, S. (2003) Ignalina accident localisation system response to maximum design basis accident. *Nucl. Energy*, **42** (2), 105–111.

EUR 20056 (2001) Main characteristics of nuclear power plants in the European Union and candidate countries, October 2001.

George, B.W. & Board, J.A. (1987) The Sizewell-B design. *Nucl. Energy*, **26** (3), 133–148.

http://www.british-energy.co.uk.

http://www.okg.se.

Handbok over processainband vid störningar I svenska kokarreaktorer, Internpumpsanläggningar, July 1987.

Hedges, K.R. (2003) The advanced CANDU reactor. *Nucl. Energy*, **42** (4), 225–229.

Hewitt, G.F. & Collier, J.G. (2000) *Introduction to Nuclear Power*, Taylor and Francis, London.

IAEA Publications, Power Reactors in Member States 1974, 1980, 1981.

IAEA TC/RER/9/004 (1994) Database for safety analysis of WWER-440 Model 213 NPPs, Vienna, Austria, August 1994.

Leclercq, J. (1986) *The Nuclear Age*, Sodel, Paris.

Lederman, L. (1995) The IAEA extra budgetary programme on the safety of WWER NPPs, *Proceedings of the International Topical Meeting on VVER Safety Programme, Czech Republic, September 21–23, 1995*.

Mounfield, P.R. (1991) *World Nuclear Power*, Routledge, London.

Nuclear Technology Review, 2003 Update, IAEA.

Ramsey, C.B. & Modarres, M. (1998) *Commercial Nuclear Power: Assuring Safety for the Future*, Wiley, Chichester, UK.

Smitton, C. (2000) Magnox history, current status and future strategy. *Nucl. Energy*, **39** (3), 171–174.

1997 World Nuclear Industry Handbook Nuclear Engineering International, Wilmington Business Publishing, 1997.

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Chapter 2

Continued Operation of Existing Plant

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Chapter 2

Continued Operation of Existing Plant

2.1. INTRODUCTION/OBJECTIVES

There are approximately 440 nuclear reactors in operation in about 30 countries worldwide. For the continuation of nuclear power, the most important requirement is the safe and efficient operation of these reactors. This chapter summarises the principal issues associated with the operation of current generation nuclear power plant. These relate to the incentives for continued nuclear generation (including its benefits as a carbon free generator), international policy, economics, safety, extension of plant life and public safety concerns. These issues are covered in more detail in separate succeeding chapters.

At the time of writing, the main focus of the nuclear industry in most countries is the continued operation of existing plant rather than on the building of new plant. This is particularly true in Europe and the US. However, some building is continuing in the Asian nuclear power states. The anticipated nuclear generating capacity at least until 2010 is expected to be comprised mainly of generation from plants in operation today (Chamberlain, 1997).

The main criteria for continued plant operation are that the plants must remain safe to the satisfaction of the regulators but also economically viable to meet the requirements of the utilities and the stakeholders. Other pre-conditions that are likely to apply to continued civil nuclear power generation in general, including new build, are separation from weapons programmes, openness and good communication of the issues and effective waste management.

2.2. INCENTIVES

This chapter examines the incentives for future energy production from nuclear (carbon free) power generation in general. The potential economic incentives for the continued operation of current generation plant are also considered. These depend on whether the costs of maintaining and renewing the plant licence (which will in general increase with life) and other generation costs remain acceptable, compared with the revenue earned by the plant and perhaps other economic factors. Other broader incentives, e.g. environmental benefits are common to both continued operation of existing plant and the building of new plant. These are considered in more detail below.

World energy supply is dominated by fossil fuels (Table 2.1). It is generally accepted around the world that there is a need to reduce the emissions of greenhouse gases from the

Table 2.1. Percentage of world energy use

Fuel	Percentage (%)	Present trends
Oil	39	Building of more fossil fuel plants
Coal	25	
Gas	22	Short-term – greater burning of oil, coal and gas resulting in more CO_2
Hydro	7	
Nuclear	6	
Renewables	1	Greater energy efficiency – increased renewable sources of energy: geothermal, wind, solar, bio-mass

Data from Blix (1998).

burning of fossil fuels. These can be reduced somewhat by increased dependence on renewables and by energy savings, but a continued or possibly increased dependence on nuclear power is likely to be the only credible option to achieve the limitations in greenhouse emissions that are thought to be necessary.

There is general agreement that there will be an increase in the world's requirement for electricity over the next few decades. The World Energy Council (WEC) (Blix, 1998) predicts that the expansion will increase by 50–70% between 1990 and 2020. The drivers are increase in world population, expansion of industry and improvements in standard of living particularly in the developing countries, e.g. Asia.

The present trend towards meeting this demand includes the building of fossil fuel plants, particularly combined cycle gas fired (CCGF) plants. There are at present no orders for new nuclear reactors in Europe or North America (Finland may place an order in the near future). There is some limited completion of plants in Eastern Europe. There are still a number of new reactors under construction in Eastern Asia. The consequences of this 'little or no nuclear build' strategy are increasingly greater emissions of carbon gases, together with other gases associated with the burning of fossil fuels (e.g. sulphur oxides).

The spiralling increase in greenhouse gas emissions has resulted in the setting of targets for many of the individual industrialised countries and international bodies concerned with nuclear energy. Although sound in principle, this approach has met with only limited success. Targets were set in Toronto (1988) to reduce emissions of carbon dioxide by 20% by 2005, in Rio (1992) to return to 1990 levels by 2000, by the UN General Assembly (1998) to achieve a 15% reduction of greenhouse gases by 2010 compared with 1990; a means of achieving constraints was put forward at the Kyoto conference (1997). In practice, however, emissions have significantly increased. From 1988 to 1998, carbon dioxide emissions have increased globally by about 16%. IAEA predict that emissions will be 36–50% higher by 2010 compared with 1990. Figure 2.1 (Energy Visions 2030 for Finland, 2003) shows past and projected carbon emissions in the industrialised and

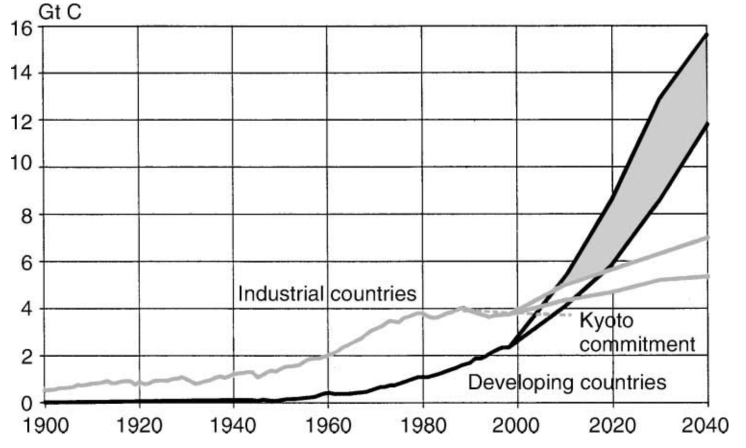

Figure 2.1. Fossil fuel carbon dioxide emissions. Source: Energy Visions 2030 for Finland (2003).

developing countries for a future scenario based on a relatively robust market development with a fossil fuel-based economy. The Kyoto protocol limit (indicated by a dotted line and applied here to the CO_2 from fossil fuels only) is also shown.

Ways have been proposed to reduce these increases by directly reducing the quantity of greenhouse gases produced, by such means as increased efficiency, via national economic constraints or by the setting of global limits that define national quotas, etc. Renewable sources of energy should not be ignored but there are technical limitations on the scales of operation that might be required, e.g. the size of wind farms. There are also issues of reliability and transmission; the wind does not blow every day and the power may be generated in remote areas or out at sea. Finally, new technologies have been proposed to convert harmful flue gases such as sulphur and nitrogen oxides to ammonium salts, by adding ammonia to flue gases and then irradiating with an electron beam produced by a nuclear accelerator. These techniques though do not apply to the carbon dioxide emissions associated with the burning of fossil fuels. Other long-term solutions, e.g. hydrogen and fusion do not provide viable alternatives on a timescale of the next few decades.

Many commentators, therefore, feel that the only viable alternative to fossil fuels is nuclear energy to reduce the rate of increase of greenhouse gases, particularly carbon dioxide.

Another incentive for nuclear power is to maintain diversity of supply. A national strategy limited to one particular form of energy (fuel) will be vulnerable to reductions of other fuel costs.

There are differences in view on the economic competitiveness of nuclear electricity compared with other fuels. Clearly, there are significant uncertainties in future costs, looking forward over a timescale of the life of a plant (at least several decades).

2.3. INTERNATIONAL POLICIES

A review of the place of nuclear power in world energy generation compared with other energy sources has been carried out by Birol (2000). The work is in the context of the International Energy Agency's World Energy Outlook (World Energy Outlook, 1998). This paper projects that nuclear energy generation worldwide will be broadly at the same level in 2020 as at present (Figure 2.2) and summarises differences in national policies. It is clear that there is marked difference of prospect across the various world sectors.

Nuclear electricity production is increasing in China, and in other developing countries and particularly in Asia (Figure 2.3). The most notable examples are Japan, Korea and Taiwan. Other countries planning expansion include India and Pakistan (Fisk, 1999). The main reason for the increased production is the building of new plants and indeed the share of nuclear electricity in these countries is increasing. Other Asian countries are also considering building. These include Indonesia, Thailand, the Philippines and Vietnam.

In North America, the situation is less certain. There could be a significant decline in nuclear generation since a number of the US plants are older reactors. However, there are increasing drives to extend the life of older plants. In recent years, there have been generally positive statements on the prospect of building new plants in the US in the future.

The situation is similar in Europe where no new plants have been ordered and relatively few plants are under construction. There is also a marked variation in national policies from country to country.

The reasons for the overall decline in Europe and North America also vary from country to country. In most countries, the reasons are partly economic and partly political. In the UK for example, there are no restrictions in principle on the building of new plant (subject to regulatory approval); the issues are primarily economic. A similar position exists in Finland. Elsewhere in the EU, Belgium, Germany, Netherlands, Spain and Sweden

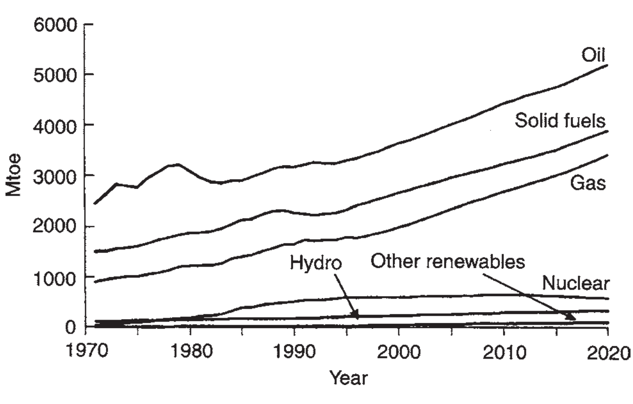

Figure 2.2. Total world energy demand. Source: Birol (2000).

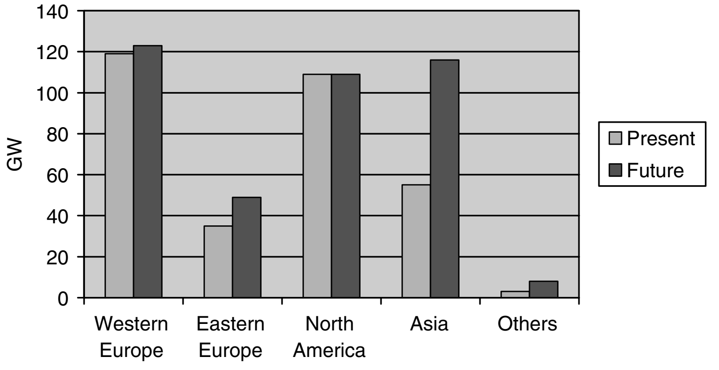

Figure 2.3. Nuclear generating capacity in 2010. Source: Chamberlain (1997).

continue to operate nuclear plant, but have a moratorium on the building of new plant. In Italy, all nuclear plants have been shutdown since 1990 and there was an immediate moratorium on building. Other EU countries, e.g. Denmark, Greece, Ireland and Norway have never built nuclear power reactors and none are foreseen in the future.

In Russia, Ukraine and the Central European Countries, there is a positive attitude to nuclear power production. A large number of reactors are in operation and dependence on nuclear power is necessary to provide these countries' energy requirements in the short to medium term. However, some of the older designed reactors built during the Soviet era are generally recognised as having safety limitations. There are political forces that these should be closed down. Many such plants have already ceased operation. New electricity producing plants are being required to fill gaps in supply and new nuclear power plants are filling that demand.

Nuclear power plants are also in operation in South America (e.g. Argentina and Brazil) and in South Africa. There is currently a global initiative by ESKOM to design a new high-temperature reactor based on an earlier pebble bed design. This reactor type is discussed later in the book. In Australia, there are no plants currently in operation and there are restrictions in place on future building.

2.4. ECONOMIC ISSUES

As noted earlier, the emphasis of the power generation industry in many sectors (e.g. Europe and North America) at the present time is the continued successful operation of existing plant, rather than the building of new plant. The main economic reason behind this position is the high capital cost of new plants, a degree of stagnation in demand, and the availability of cheap gas.

Table 2.2. Average generation costs ($US per kWh)

Generator	5% Discount	10% Discount
Nuclear	0.034	0.051
Coal	0.038	0.048
Gas	0.040	0.044

Data from Wilmer and Bertel (2000).

Operation of current plant will continue provided that they remain not only economic but also safe and environmentally compliant with regulatory guidelines. To be economic, a number of factors need to be considered and these are sector dependent. The plant may need to operate in a de-regulated market in competition with other generators. The economics will depend on electricity prices, which may be reducing, and also on other operating costs. These issues are discussed below. Clearly, overall costs have to be achieved against budget and kept to a minimum, without compromising safety of the plant.

The economic competitiveness of nuclear power plants has been the subject of several OECD studies (Wilmer and Bertel, 2000). These studies have analysed the projected costs of generating electricity compared with alternatives. In Wilmer and Bertel (2000), 'levelled cost comparisons' are presented from 12 countries, each providing information for at least one nuclear unit and one alternative. The costs were calculated making common assumptions. For nuclear plants, these included a 40-year lifetime and a 75% load factor. For gas-fired plants, the assumptions included the cost of replacing major equipment after around 20 years. The costs were levelised at 1997 costs.

Nuclear generating costs are the most sensitive to discount rates. The above Table 2.2 indicates that at the time of the study, nuclear power is competitive at 5% discount rates but loses its competitive margin at 10%.

2.4.1 Fuel Costs

Fuel purchase costs for nuclear plants are generally low in comparison with other energy producers (Table 2.3). These costs are an important differentiator to the competitiveness of nuclear vs. non-nuclear plant. Nevertheless, fuel purchase costs are still high and can significantly affect the economics of plant operation.

Table 2.3. Fuel costs

Generator	Cost (% of generation)	Comment
Nuclear	<25	Relatively insensitive to uranium price volatility
Coal	~40–50	Sensitivity to coal price volatility
Gas	~75–80	Very sensitive to gas price volatility

Wilmer and Bertel (2000).

For a nuclear power plant, over half of the generating costs relate to the initial capital investment. Fuel accounts for less than 25% of the total generation cost and in recent years, fuel cycle costs have decreased significantly in all countries. Conversely for coal and gas, fuel costs are the most dominant, representing 40–80%, respectively, of the total generation cost. Regarding other costs, operating and maintenance (O & M) costs represent only a small part of the total generating costs of nuclear power plants. These O & M costs relate mainly to the technical performance of the plants, safety regulations and staff costs. Decommissioning costs' issues are discussed later in Chapter 6.

2.4.2 Safety Upgrades' Costs

The costs of safety upgrades have been considered in IAEA-TECDOC-1084 (1999). In this section, the costs for continued operation within the design life of the plant are considered specifically, costs associated with plant life extension (and decommissioning) are considered later in Chapter 6. It is also recognised that it is not usually possible to separate out from the available data, the costs associated with plant performance or for normal equipment replacement, against the costs associated with an actual safety upgrade.

In IAEA-TECDOC-1084 (1999), costs (levelised to 1997) associated with a range of water reactor types are reviewed. These include PWRs and BWRs from the US, Korea and Western Europe (Germany and The Netherlands) and Russian-designed VVER and RBMK plants in Central and Eastern Europe and the Russian Federation.

The cost estimates per unit of plant capacity and per year were considered for different categories of plant age (in 3-year period spans) for both PWRs and BWRs (Table 2.4). Costs over 5 years are also given to enable broad comparisons to be made against VVER and RBMK data covering costs of safety upgrades on these plants, carried out over the last few years. The average figure for BWRs was somewhat higher than PWRs (Table 2.4; IAEA-TECDOC-1084, 1999). However, it was concluded that the costs were not particularly reactor dependent.

It was also found that costs of upgrades depended on the age of the unit. In the first few years, costs were relatively high associated with bringing units up to latest regulations; this was followed by a period of lower costs; costs then started to rise as ageing factors start to become an issue.

Assessments for the Russian-designed VVER series were also carried out (IAEA-TECDOC-1084, 1999); reference data are shown in Table 2.5. The VVER-440/230 design

Table 2.4. US Safety upgrade costs ($US per kWe)

Plant	Estimated costs/year	Costs over 5 years
PWR	27	135
BWR	32	160

Data from IAEA-TECDOC-1084 (1999). Assumptions – average for different plant age categories.

Table 2.5. VVERs: Safety upgrade costs ($US per kWe)

Plant	Estimated costs	Generation of VVER
440/230	70–162	1st
440/213	23–34	2nd
1000	17–31 (Russia) 201–277 (Ukraine)	3rd

Data from IAEA-TECDOC-1084 (1999).

has recognised deficiencies in relation to the integrity of the reactor vessel, the confinement pressurisation limit, and the limited scope of design basis accidents. The VVER-440/213 contains safety enhancements compared with the 440/230, particularly in terms of enhanced confinement capability, extended design basis for pipe breaks and more safety system equipment redundancy. This is reflected in the costs of the safety upgrades for VVER-440/213s, being 2–3 times lower than those for VVER-440/230s.

The VVER-1000s are better equipped again with a stronger containment. Additional enhancements have also been identified to achieve improvement of core behaviour and measures introduced to protect the integrity of the steam generators. Thus for VVER-1000s, the costs are still relatively high. The Russian Federation estimates were much lower than the corresponding Bulgarian and Ukrainian estimates. It is worth noting that safety enhancements were implemented earlier in the VVER-440s because of the perceived urgency. The modifications of the VVER-1000s were of lower priority.

Important areas for safety enhancements of RBMKs have been identified, e.g. reduction of positive steam reactivity coefficient and improvement of the scram systems. It is clear that RBMK reactors still require investment of at least the same order as other plants, although a large part of the investments has already been made. Data are shown in Table 2.6.

The OECD study concluded that a new nuclear plant is unlikely to be the cheapest option, but that existing nuclear power plants provided they were operated and well managed can have a clear economic advantage, because of their low marginal costs. An important factor for the economic equation is whether a plant can operate reliably and in a stable condition, i.e. achieve a high load factor. Efficiency and performance are considered in Chapter 4. The load factors of nuclear plants tend to be less than those of fossil fuel plants.

Table 2.6. RBMKs: Safety upgrade costs ($US per kWe)

Plant	Estimated costs	Comments
1000	38–97 (Russia)	Investment already made
1500	76–125 (Lithuania)	

Data from IAEA-TECDOC-1084 (1999).

The economic benefits of continued operation of the Magnox plants in the UK have been published (Mortin, 2000). The UK electricity market is de-regulated and the Magnox stations have to compete with the other electricity generators. The price for electricity in the UK market has also reduced in recent years. Nevertheless Magnox stations have continued to operate for many years and some will continue to do so for the next decade. In the main, Magnox stations have achieved very respectable load factors. Nevertheless, the marginal contribution from individual Magnox stations of the fuel purchase costs is substantial and stations are closing. Fuel costs are increasing as there is a diminishing requirement for metallic fuel, as more stations are shut down (Smitton, 1999, 2000; May, 2003).

However, in summary, there are economic benefits in continuing to operate many existing nuclear plants. At the present time, these benefits are being further realised with the help of good management providing efficient, cost effective measures for running and ensuring the safety of the plant.

2.5. SAFETY OF OLDER PLANTS

As far as possible, there is a need that all plants take into account developments in safety standards and technology. It is unlikely that older plants will meet the same standards as modern plants but they must have adequate operating safety margins. These are assessed, for e.g. by following plant modifications, new fuel cycles and also during periodic safety reviews (PSRs), discussed in more detail later in the book. Most utilities are required by their regulators to carry out PSRs at regular intervals, typically at least every 10 years. The purpose of these reviews is to consider all facets of the long-term operation of the plants (rather than the particular every day running of the plant).

Considerable experience has been gained in the UK on the continued operation of nuclear power plants over the past 50 years. Some particular activities that are being carried out in support of the Magnox reactor programme (Mortin, 2000) are described below and are typical of the practices that need to be adopted for older generation plants. These are summarised in Table 2.7.

Plant maintenance and monitoring practices must be reviewed to take advantage of improved techniques. Particular checks must be made on the functional testing of

Table 2.7. Important issues for continued operation

Increased safety demands – impact of new standards and technology on performance and operation
Plant maintenance and monitoring – availability of improved techniques
Ageing – status of plant and how undesirable effects can be mitigated
Long-term technical support – availability of Suitably Qualified and Experienced (SQEP) personnel

Mortin (2000).

components and on equipment settings. If necessary, components under wear must be replaced. Another purpose of the reviews is to establish that the frequency and scope of inspections is optimal from the point of view of both safety and cost effectiveness.

The issues of structural plant ageing have to be addressed. Plant ageing can result from many wide-ranging and different phenomena depending on the plant in question. Pressure vessels may become embrittled as a consequence of high fluence particularly at welds. Chemistry effects such as oxidation of graphite cores is a particular issue in gas reactors that needs to be considered.

Another issue identified concerns the technical support of plant. There are issues arising from the potential loss of staff, perhaps recruited in the early days of plant operation but who may be nearing retirement several decades later. The problems of recruiting into a nuclear industry that may be scaling down are well recognised.

2.6. ENVIRONMENTAL ISSUES

Nuclear power plant operation along with many other industrial plant operations is inextricably linked with a number of environmental issues. These have and are being considered within a global context, e.g. within the UN (Stockholm 1972 and Rio 1992) and also within the EC. National governments are also addressing these issues by charging various government bodies, agencies and commissions to advise on policy and propose discharge consents, etc. to meet national and/or international targets for emissions.

In the UK (Fisk, 1999), a number of enquiries into future nuclear power have addressed environmental issues in their deliberations. For example, within the past few years the House of Lords has conducted an enquiry into nuclear waste, the Environmental Agency has proposed discharge consents for reprocessing at Sellafield and the Royal Commission for Environmental Pollution has considered evidence on energy and the environment. These initiatives have largely been driven to make input to the debate on how the UK can meet greenhouse gas emission targets for the period 2008–2012. This issue has been a consideration in the UK Energy Strategy Review (Performance Innovation Unit, 2002; DTI Energy White Paper, 2003). The greenhouse gas targets are particularly challenging. Figures 2.4 and 2.5 show the dependence on nuclear energy in 2002 with nuclear electricity representing 23% of the total electricity supply. Without new building, the nuclear fraction figure will reduce with the shutting down of all the remaining Magnox stations by 2010, and some of the remaining AGRs by 2020 (Table 2.8).

Fisk (1999) considers environmental issues within the wider context of ‘sustainable development’ to which the UK government is committed. There are a number of definitions of this concept. A common definition is ‘meeting the needs of our generation without compromising the ability of future generations to meet their needs.’ This definition was put forward by the Brundtland at the end of the 1980s. Generally, the term has come to

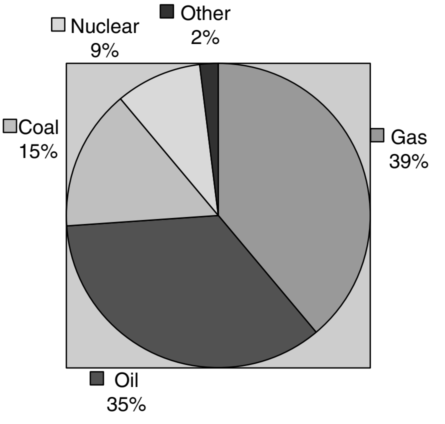

Figure 2.4. UK primary fuel mix in 2002. Source: Digest of UK Energy Statistics (2002).

mean improved welfare for everyone both in the present and the future. The key concept here is 'improvement for *everyone* as opposed to improvement for *some* at the expense of others.'

Sustainable development and nuclear power invoke a number of issues, perhaps the most important is the issue of waste. Nuclear waste can in principle cause harm to future generations, which could certainly result in the future without an adequate waste strategy. Since in most countries at the present time, there is no agreed strategy, the question must be asked whether it is justifiable to continue with nuclear energy power production, thus generating waste in the hope that future generations will be able to solve the problem.

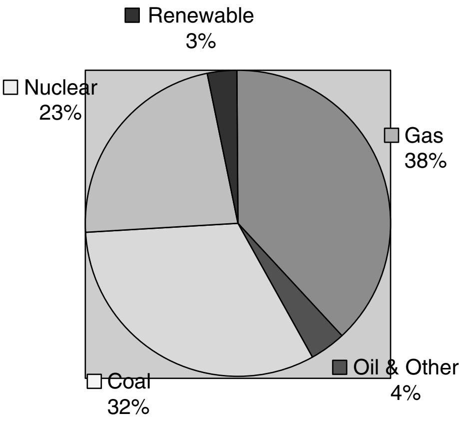

Figure 2.5. UK electricity generation in 2002. Source: Digest of UK Energy Statistics (2002).

Table 2.8. Projected rundown of UK nuclear stations

Station	Commissioning date	Status	Closure date
Bradwell	1962	Shutdown	2002
Calder Hall	1958	Shutdown	2003
Chapelcross	1959	Operational	2005[a]
Dungeness A	1966	Operational	2006[a]
Dungeness B	1983	Operational	2008[a]
Hartlepool	1983	Operational	2014[a]
Heysham 1	1983	Operational	2014[a]
Heysham 2	1988	Operational	2023[a]
Hinkley B	1976	Operational	2011[a]
Hunterston B	1976	Operational	2011[a]
Oldbury	1968	Operational	2008[a]
Sizewell A	1966	Operational	2006[a]
Sizewell B	1995	Operational	2029[a]
Torness	1988	Operational	2023[a]
Wylfa	1971	Operational	2010[a]

Data from Mayson (2003).
[a]Denotes projected date.

Different countries may have different levels of acceptability. There are legacies of inadequate waste disposal in some countries that are now posing significant problems (and expense) to resolve. Practices have been adopted that would now not be considered as acceptable, yet were considered so at the time. Thus, levels of acceptability can and do vary from one country to another and will also change with time. Further there may be economic reasons to transport waste from one country to another, perhaps with less stringent environmental standards. Is this acceptable, both from a global environmental standpoint, or indeed from a moral standpoint – clearly the answer should be no.

The liabilities associated with decommissioning nuclear power plants once they have reached end of life are another important issue. These are obviously inescapable for currently operating plant; liabilities are a critical factor with regard to decision-making for the building of new plant. Having adequate decommissioning plans prior to building is now typically a regulatory requirement. In the UK, for example, there must be such a provision. From an economic perspective, there is the issue of whether adequate funds are in place for decommissioning, these may be available through increased price levies, or government underwriting of liabilities.

Aside from the concerns of waste disposal, perhaps the major environmental requirement from the public is that the risk of severe accidents is not only small, but also that if such accidents were to occur they can be effectively managed. This is a particular concern for older currently operating plant, which perhaps (and indeed were) licensed under more tolerant licensing regimes than those of the present day. Such plants may have greater vulnerabilities for severe accidents than some modern plants.

It has been discussed above that an important environmental benefit put forward for continued nuclear power plant operation is that it does not contribute to increased global warming and acid rain. The challenge is how to realise this benefit. National governments may set up infrastructures offering incentives to reduce emissions, which might be achieved either by building of new nuclear plant or through life extension of existing plant. In the latter case though, safety must not be compromised. Pressures to continue operation may be very great where no alternative power producers are available. This may be particularly true in the less developed countries. From an environment perspective, clearly safety considerations and the avoidance of adverse environmental consequences resulting from an accident must be paramount.

2.7. NUCLEAR COMPETENCE

At the present time, there is a general decline in many areas of support for the nuclear industry. Research and Development programmes have been particularly affected, certainly in the Western world. The reasons for this are that the knowledge base to support currently operating plant is at a relatively mature state and the lack of new building programmes means that little new work is needed. In addition, nuclear energy is having to compete with other forms of energy producers in the market. The nuclear business is not seen as a popular business in which to work. The net result is a significant reduction in resource due to a failure to attract new graduates into the industry, a failure to keep new people in the industry and the loss of people in retirement.

It has been recognised for some time therefore, (Storey, 2001) that there is a requirement to maintain technical competence, not only to ensure safe operation and decommissioning of existing plant, but also to be available in the future, if new reactors are required. The continued operation of existing plant does provide a means to ensure some level of competent resource is maintained for both operation and regulation.

Regulators are focussing on a number of areas through the NEA described by Storey (2001). Specific problems have been considered by the Committee on the Safety of Nuclear Installations (CSNI)'s Senior Expert Group and more recently by the European Commission through its research and training programme (RTP) in the field of nuclear energy. The Senior Expert Group has made recommendations for research in a number of important technical areas in the OECD Community. These include the maintenance of a major thermal–hydraulic rig for each reactor type, for fuel and reactor physics facilities, for research on the integrity of equipment structures and for the continued availability of hot cell and research reactor facilities. EC initiatives include the creation of centres of excellence (COEs) for severe accidents and for fission products expertise; the setting-up of databases on seismic activity and support of other areas in respect of human factors and plant monitoring and control.

Other initiatives are moving forward under the auspices of the NEA Committee of the Nuclear Regulatory Authorities (CNRAs), which are more general (including non-technical topics). Some of these are aimed at maintaining safety competence in the industry and the regulator. The NEA and the EC in its RTP programme, referred to above, are also addressing nuclear training and education.

Thus, maintaining a sufficient degree of overall competence is a particular issue in the nuclear industry at the present time (BNIF/BNES Conference – Energy Choices, 2002). As nuclear power is declining internationally and particularly the lack of 'new build', there are problems with the retirement of suitably experienced and qualified (SQEP) staff and difficulties in recruiting high-quality personnel into the nuclear industry. In countries where there are continuing nuclear programmes, there is at least a steady stream of work to support plant operation so some capability is maintained.

2.8. EXTENSION OF LIFE

It is likely that nuclear power within particular sectors will decline over the next 20 years. However, increasing competition will encourage utilities to seek plant life extensions, tending to slow this decline and contribute to reducing carbon dioxide emissions. It is probable that with appropriate investment and refurbishment, the lives of some plants may extend up to 60 years and beyond.

Many present-day reactors are now approaching the end of their design life. There are considerable efforts to extend the operational life of such plants by various means such as backfitting of systems, changes in operational practices, etc. For many countries, the economics of extending the life of existing plants, compared with the capital costs of building new plant, is very favourable. However, the Chernobyl accident in particular has shown that reactor safety is an international concern and economic benefits have to be considered against global acceptability. Decisions on the extension of life depend on a range of technical issues, principally materials performance, chemistry and availability of sophisticated inspection techniques. These and other more general issues (Table 2.9) are reviewed in this section.

Table 2.9. Extension of lifetime issues

Technical feasibility – effect of the processes of ageing?
Plant safety for intended period of operation – ageing of critical safety components?
Regulatory framework – establishment of procedures for licence extension?
Social acceptability in national climate – changes during plant lifetime and public perception?
Economic considerations – are the economics favourable?

IAEA-TECDOC-1084 (1999).

2.8.1 Operational Limits

There are many reactors in operation over the age of 25 years, relative to typical licensed operation of 30–40 years (IAEA-TECDOC-1084, 1999; Figure 2.6). Without extension of life there would need to be significant investment to replacing generating capacity by new plant (nuclear or non-nuclear). This is the situation in many countries in Western and Eastern Europe, in the Russian Federation and in the US.

There are obvious incentives in extending the life of a plant. The capital costs of building new plant (even non-nuclear plant) are likely to be high compared with a plant that is continuing to operate well. Decommissioning activities will be delayed and thus present day decommissioning costs are avoided. In decision-making for lifetime extension, there are various factors that need to be considered.

Firstly, the technical feasibility must be considered. The performance of major plant components for an extended period must be guaranteed. Integrity of structures such as the reactor pressure vessel, steam generators, pressurisers, primary and secondary circuit pipework and the containment structures must be assessed against any deleterious effects of ageing. In general, ageing processes reduce margins for operation and it may be difficult to substantiate the performance of these components for long lifetime extensions since the original lifetime of the plant will have been set at the operational limits of these key components in the first place.

2.8.2 Safety Issues

In order to extend the operating life of a reactor, it is necessary to review all aspects of the safety case. Plants are usually designed for a certain life, which is based on knowledge at the time and forecasts extending over a period of 20 years or more. In practice, the actual working life may be different and depend on a number of factors. These might include (Twidale, 1999):

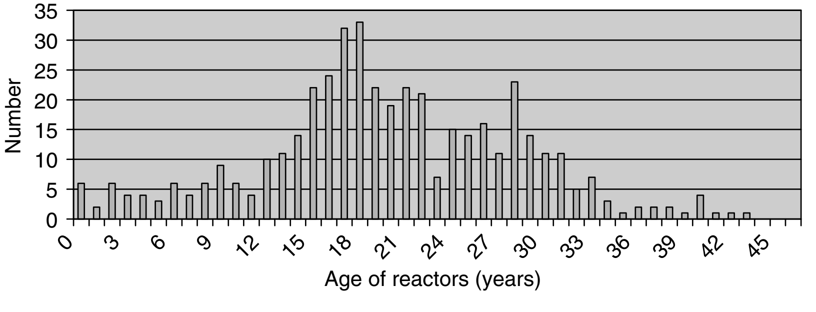

Figure 2.6. Age distribution of operating reactors in December 2002. Source: IAEA Technology Annual Report (2002).

- changes in operating conditions compared with the assumption in the design (these could affect margins);
- findings from maintenance inspections;
- results of test programmes and;
- outcome of safety assessments.

In addition, the operating experience of the plant (and possibly sister plants) and the accumulation of materials and other plant data will also impact the life. If these are favourable, a licensee may seek permission from his regulator for life extension.

As discussed earlier, PSRs have been introduced as a means of reviewing the safety of a plant on a regular basis. The results of PSRs strongly influence decisions for future plant operation and become increasingly important for older reactors or plants where life extension is under consideration. PSRs are conducted, usually at least every 10 years; for some plants they are conducted more frequently, particularly during later life.

The case for plant life extension would have to confirm the plant's safety for the proposed additional operation. It would include identifying any features that might restrict the plant-operating envelope during this period. A secondary objective may be to assess the plant's safety standards against current safety standards. This objective is usually realised somewhat partially since it is realised that it is not reasonable to expect older designed plant to wholly meet the safety standards of the day. In this circumstance, consideration would be given to the age of the plant and the intended life extension.

The top priority components for safety justification review are likely to include the reactor pressure vessel, control rod drive mechanisms, internals, supports and the biological shield, the primary circuit including the pressuriser and steam generators, coolant pumps, containment structures, and control and instrumentation. Other more minor components clearly have a safety justification, e.g. valves, smaller pumps, sensors, etc. but these components would have been routinely replaced under regular operation within the original design life.

2.8.3 Regulation

Clearly decisions on granting life extension will rest with the regulator of the country in question. He will need to have established procedures in place and if not already available, these will take time to develop. There are now precedents in a number of countries on regulators considering or having already granted applications. For example, in the UK, the NII have already extended the lifetime of some of the Magnox reactors beyond 40 years. Several US utilities have applied for lifetime extension. The issue is now under consideration in Canada, Japan and European countries, including Russia (for some designs).

2.8.4 Political Factors

Undoubtedly, the climate of acceptability of nuclear power has changed during the lifetime of plants in many countries. Many plants commenced operation when the view that 'nuclear power would be too cheap to meter' was being expounded and attitudes towards nuclear power were very positive. Now, decades later there may be a moratorium in certain countries on extending plant operation beyond the original operating life. Nevertheless, there are a number of countries that are not opposed to the continued operation of nuclear power plant and the decision will then become one of economics and safety compliance.

2.8.5 Factors affecting the Economic Decision

The economic case will depend on many factors. The principles for estimating the lifetime extension costs are not different from those used for assessing the costs of plant upgrades, design changes, plant decommissioning or new construction. However, there may be some particular site-dependent factors that need to be taken into consideration.

Plants built during the early days of nuclear power production tended to be diverse; as lessons were learned, designs were improved, and unit capacities were increased. Thus, applications and therefore costs for life extension for early designs are likely to be design specific. This will be less so in the future as later current generation plants reach their design end of life and are considered for life extension.

Even for a given reactor design, there may be significant technical differences in the state of the important components. An obvious example is the state of the reactor pressure vessel. This will largely depend on the integrated neutron fluence experienced during its design life. This will be affected by the power level at which the plant has been operated, the effectiveness of the vessel protection radiation shield, whether the vessel has been annealed, etc. In short, the operating history will be needed to assess the state of the vessel.

Measures may be taken to upgrade certain components during initial planned life. These may be upgrades to improve performance, improve safety, or to improve or mitigate the effects of ageing. For example, steam generator replacement has been performed on a number of plants to improve reliability of operation. Component replacement during normal design life will clearly be beneficial to improving the chances of life extension for the particular plant. Further, higher costs incurred during normal operation could well mean reduced costs for life extension.

Other factors identified in IAEA-TECDOC-1084 (1999) include differences in costs due to differences in regulations between countries, resulting in differences in costs in the safety cases presented by the utilities. Another factor in assessing the comparative economics of lifetime extension costs rests with the proprietary nature of plant data. Utilities are reluctant to release information that might benefit another competitive utility.

Generic cost data for US plants have been published in IAEA-TECDOC-1084 (1999). These show lifetime extension costs relative to the building of new nuclear plant compared

Table 2.10. Lifetime extension versus new building costs ($US per kWe)

Lifetime extension	210–840	Based on Surry 1 (PWR) and Monticello (BWR)
New nuclear plants	≅ 2000	Building costs only
New combined cycle units	700–900	Building costs only

Data from IAEA-TECDOC-1084 (1999).

with combined cycle plants. There are considerable uncertainties on life extension costs but the conclusion is favourable for life extension at least on the basis of building costs of new plant (Table 2.10).

Also in IAEA-TECDOC-1084 (1999), an attempt is made to identify site/plant specific factors that have most influence on the cost and tend to push the cost estimates to either the higher pessimistic or to the lower optimistic figure.

With regard to plant design, steam generator replacement and reactor pressure vessel annealing were key factors for PWRs while replacement of pipes and reactor pressure vessel internals were the key factors for BWRs. For reasons explained earlier, newer plants would be expected to incur lower costs than older plants. However, this is somewhat obscured since the former will tend to have longer design life than the latter.

The schedule for implementing the various measures is important. If the intention is to continue the extended operation immediately beyond the end of design life, it is advantageous to begin lifetime extension measures during earlier scheduled outages. Other costs identified were the costs of replacement power to meet the demand during the intervening period. Such costs are clearly highly power system specific.

Finally, costs to meet the demands of the regulator and also possibly to overcome the concerns of the public also need to be factored into the cost balance.

It is concluded that plant-specific costs be required in order to make realistic cost estimations. To evaluate the competitiveness of life extension options, it will be necessary to compare with other power-producing options, including both nuclear and non-nuclear. However, while not possible to produce a generic economic case for life extension, it is clear that a number of utilities have addressed the issue for their own plants and have come out in favour of the benefits of life extension.

2.8.6 Specific Examples

Finally in this section, two examples of life extension proposals are cited for illustrative purposes.

2.8.6.1 Magnox Stations. In the UK, safety cases were assembled to extend the operating life of the Magnox reactors up to 40 years (Twidale, 1999). The original design life of these plants was 20–25 years. These plants were commissioned in the 1960s and 70s

Table 2.11. Issues for Magnox plant for extended operation

Degradation due to ageing – ageing of civil structures
Hot gas release – evaluation of gas ducts' failures on post-trip cooling
Seismic events – evaluation of risk, plant integrity and ALARP principle
Extreme wind – engineering assessments of RPV and cooling pond foundations
Extreme flood – assessment of groundwater level extreme changes
Design codes and standards changes – structural modifications where required

Twidale (1999).

and many are still in operation today. The last station, Wylfa, is not due to shutdown until 2010.

Issues of concern in the cases covered both hardware and software, together with plant and system reliabilities and key performance parameters such as load and temperature histories (Bolton, 1996; Table 2.11). The hardware concerns covered the ageing of any materials that might result in loss of structural integrity and how these could be inspected and also electrical hardware. Software issues concerned the demonstration of safety margins and system reliabilities in terms of economic performance were also evaluated.

As a consequence, a programme of long term safety reviews (LTSRs) was instigated in the early 1990s to be followed up with PSRs at least every 10 years up until the end of each station's life. These PSRs in some cases require more frequent inspection of some key components than every 10 years.

The scope of the Magnox PSRs covered re-assessment of the remaining safety margins in the plant, using probabilistic methods, where possible. Issues considered included: ageing degradation of the structures, fault loadings, external hazards (e.g. seismic, wind, flood, etc.) and changes to design codes and standards. Where structural ageing had occurred, modifications or repairs were carried out to ensure structural integrity was maintained. Studies were carried out to confirm the plant's safety level for fault loadings and external hazards.

In particular, for Magnox plants, ageing degradation could occur due to one of a number of processes including carbonation, chloride attack (for coastal plants), and thermal and movement effects. In general though, the Magnox structures have remained in good condition. Loadings on the structures under low probability pipework failures outside the vessel have been assessed; in addition post-trip cooling of the reactor has also been demonstrated. Seismic qualification was performed taking advantage of more modern analytical 'finite element' methods than were available in the original design phase. Similarly, engineering assessments of the reactor vessel and cooling pond foundations were carried out using more up-to-date mathematical modelling and taking advantage of better understanding of the soil mechanics.

2.8.6.2 Kori PWR. A cost review has been conducted (IAEA-TECDOC-1084, 1999) for Kori NPP Unit 1 PWR, which is a 2 loop PWR that commenced operation in 1978. The original licence was for 30 years, expiring in 2008.

Table 2.12. Costs of lifetime extension of Kori-1 ($US per kWe)

Period of extension (years)	Overnight cost
10	228–361
20	433–567
30	639–772

Data from IAEA-TECDOC-1084 (1999).

A feasibility study was carried out to identify critical components from the point of view of continued operation. For the review, 13 critical components were identified (excluding the steam generators replaced in 1998): reactor pressure vessel, reactor vessel internals, control rod drive mechanisms, pressuriser, reactor coolant system piping, reactor coolant pump, reactor pressure vessel supports, pressuriser nozzles, turbine, cables, containment building and generator.

Three duration periods were considered (10, 20, and 30 years) for different implementation options. Option 1 assumed that only refurbishment and replacement of components would be needed while Option 2 included additional costs for safety back-fitting (in the area of fire protection, equipment qualification), and to withstand station black-out and the costs of licensing.

It was found that the Korean forecasts corresponded well with the US data referred to earlier. As for the US figures, the lower cost option would be competitive against best competitors, e.g. combined cycle units, but the higher cost option may turn out to be non-competitive. In Table 2.12, the lower cost figure is for Option 1, the higher figure for Option 2. However in all cases, the benefit/cost ratios were estimated to be greater than unity and therefore the lifetime extension options were all viable.

2.9. PUBLIC SAFETY CONCERNS

In addition to technical and economic factors, there are undoubtedly political issues surrounding the future operation of nuclear plant There is the issue of public or stakeholder confidence. There is not simply the question of the safety of nuclear reactors where only a few significant accidents have occurred, but no longer can the nuclear industry claim that severe accidents are incredible. In regard to normal operation, there are also public concerns on waste management issues, about the fuel cycle and on the issue of proliferation. These issues are reviewed in this section.

2.9.1 Reactor Accidents

Reactor Accidents (or the potential for accidents) are undoubtedly a public concern for the continued operation of the civil nuclear power programme. There have been comparatively few serious accidents but those that have occurred, e.g. Three Mile Island-Unit 2 (TMI-2) and Chernobyl have had a pronounced affect on the expansion (or lack of it) in the nuclear developed countries.

A good review of the accident record of the nuclear industry is given by Mounfield (1991). Some incidents have occurred during the handling of industrial isotopes or other exposures to ionising radiation; these have resulted in a small number of fatalities. Some accidents have happened at experimental facilities; e.g. a criticality accident occurred in 1961 in a small prototype BWR reactor (SL-1) located at Idaho in the US resulting in the death of three technicians. Other accidents have been recorded in experimental and power reactors involving criticality and also fuel melting. A partial fuel meltdown occurred at St Laurent 1, a 480 MWe plant in France in 1969. Incidents have taken place at San Onofre 1 in California in 1973, Brown's Ferry, Alabama in 1975 and other more minor (but still serious) events have occurred on some other plants.

In the UK, the only serious accident that caused public concern was the Windscale fire. This fire resulted in significant releases of radioactivity; estimates of 20,000 Ci of I-311 are given by Mortin (2000). As a consequence of this accident, 14 workers at the plant received serious doses of radiation, and there was a suspension of milk production in the surrounding area.

The first most damaging event in terms of limiting nuclear power plant expansion was the TMI-2 accident in 1979. The accident resulted, partly through mismanagement, in a severe core melt down that threatened the integrity of pressure vessel boundary. TMI-2 had various important consequences. It effectively terminated the construction of new power plants in the US (Chung, 1998). It also impacted on the philosophy of approach to severe accident safety. Previously, attention had focussed almost entirely on prevention, after TMI-2 there was a much increased focus on accident mitigation. Nevertheless, despite the substantial core melting, the only significant releases to the public resulted from Xenon-133 and the health consequences were not judged to be significant.

A study was carried out (Blee, 2001) to look at lessons learned over the last 22 years since the TMI-2 accident (Table 2.13).

Table 2.13. Lessons learned from the post-TMI-2 and Chernobyl era

Along with safe operations and good economics, effective communication is vital, particularly in the aftermath of abnormal events
Industry fortunes are global as further demonstrated by Chernobyl – crisis management is vital
Environmental linkages have yet to embraced – the beneficial role of nuclear energy in protecting the environment should be proposed
The need to manage media publicity
Recognition of the benefits of long-term vision

Blee (2001).

The World's worst nuclear power plant accident occurred at Chernobyl in 1986. The consequences of this accident have been much discussed and publicised. This accident resulted in a massive explosion, dispersing radioactivity over much of Northern Europe. The cause was essentially operator error but subsequent investigation indicated major weaknesses in both technical specifications and management. The Chernobyl accident resulted in moratoria for the construction of nuclear plants in some European Countries, e.g. Italy.

A relatively recent incident involving fatalities occurred in 1999 at the Tokai-mura uranium processing plant in Japan (Suzuki, 2000). This accident resulted from a fission reaction in a precipitation tank of uranyl nitrate solution. Several workers suffered severe radiation sickness ultimately resulting in their deaths, several months after the incident. This accident was investigated in depth and various deficiencies in operating processes (operational and technical, management and control), in the licensing process and in the safety regulations were identified.

As with most industries, experience from accidents has resulted in the implementation of better operational practices and technical improvements. Many of the accidents to date have resulted from human error and the need for improved training and understanding of 'human factors' issues is one of the most significant lessons learned.

2.9.2 Radioactive Waste Concerns

Although much progress has been made on the technical issues associated with waste disposal, the public has considerable concerns over the issues of waste management and the management of spent nuclear fuel (Ryhanen, 1996). As a consequence, these concerns are still some of the important reasons put forward against the building of new power plants and the sustainability of nuclear energy. Perhaps the major concerns of the public relate to the legacy of long-term radioactive waste and our obligations to future generations.

Research carried out by Duncan (2003) for the UK, Switzerland and Japan indicates that when considering the environment and family about 60% of the populations sampled considered timescales of 50 years or less and about 85% selected 100 years or less. These are much shorter timescales than are required for the isolation of hazardous wastes before their radiotoxicity is reduced, as illustrated in Figure 2.7.

The issues and status of present day approaches to waste management are considered in detail in Chapter 6.

2.9.3 Fuel Reprocessing

Reprocessing of nuclear fuel has been carried out for a number of years in various reprocessing facilities in many of the main nuclear power-producing countries. These facilities include the reprocessing plants at La Hague in France, the Tokai plant in Japan and Sellafield in the UK. A pilot study plant operated near Karlsruhe in Germany,

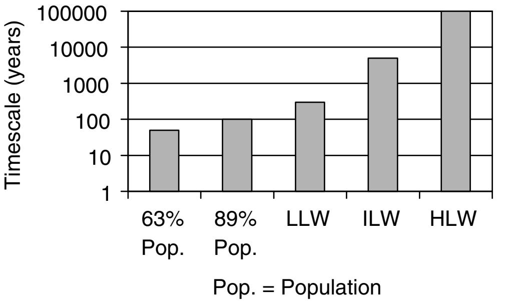

Figure 2.7. Public forward thinking and waste isolation timescales. Source: Duncan (2003).

a multi-national plant was built at Mol in Belgium and operated from 1966 to 1974 and the West Valley facility was operated for a short time in the US. Fuel reprocessing activities have also taken place in Russia, e.g. the Chelyabinsk plant, see for example Rougeau (1997).

Over the years many thousand tons of spent fuel have been processed. The plants have been adapted to take account of different types of fuel from different types of reactors. These included early gas-cooled reactors in Europe, Magnox and AGRs in the UK, most particularly light water reactors in operation in many countries and some fast reactors.

For example from 1966 to 1987, the UP2 facility at La Hague processed gas-cooled reactor fuels. Fast reactor fuel was processed between 1979 and 1984. Since 1987, UP2 has been utilised in reprocessing LWR fuels only. MOX fuel has been reprocessed since 1992. A newer facility UP3 started in 1990, having the capacity for servicing a range of spent fuels from European and Japanese facilities.

Concerning management of waste, there are currently three industrial-scale vitrification facilities in operation in Europe. COGEMA operates the R7 and T7 facilities at La Hague and BNFL operates a plant of similar design in Sellafield. Much progress has been made in reducing the volume of high- and intermediate-level waste and also in reducing the radiological dose rate to workers, see, e.g. Table 2.14.

Despite these successful operations, public safety concerns do exist and are focussed on perhaps two main issues. These include the level of release of toxic and radioactive releases from the plants and also concerns about proliferation.

Table 2.14. Reprocessing operations at La Hague

Improvement measure	1980	1995
Vol. of HLW and ILW ($m^3\ t^{-1}$ heavy metal)	3	0.5
Average worker exposure (mSv per year)	3	0.2
β and γ releases (TBq t^{-1} reprocessed)	8.87	0.03

Data from Rougeau (1997).

Regarding releases, there are concerns that small traces of radioactive material are released even during normal reprocessing operations. The public is not convinced that neither possible routes back to the public and the food chain are properly understood nor that the limits set by the radiological protection bodies are proven to be safe. The concerns in regard to proliferation are that reprocessing enables the recovery of plutonium, which could be utilised for nuclear weapons. It is worth noting, however, that plutonium recovered from LWRs (the most widely used reactor in operation today) is not in the most suitable form for weapons production.

One of the main drivers for reprocessing plutonium is to support a fast reactor fuel cycle, but only a few fast reactors remain in operation at the present time. However, reprocessed plutonium can be used in fresh MOX fuel. Plutonium only remains in a separated state for a relatively short time during the fabrication of MOX fuel, once in the reactor the fissile plutonium content is substantially lessened and a relatively high percentage (30%) of the plutonium content is burned.

Reprocessing activities are carefully monitored by the IAEA and other national bodies to ensure that proliferation issues are properly covered. It should also be stated that without reprocessing, the quantities of plutonium produced in-reactor will remain the same for many years (hundreds of years) after which time separation becomes easier following the decay of shorter lived isotopes.

2.9.4 Proliferation

The proliferation of plutonium for nuclear weapons purposes is a public concern. Significant quantities of plutonium were present in nuclear arsenals of countries with a nuclear weapons' capability, particularly, e.g. in the US and the former Soviet Union. Plutonium is managed in nuclear fuel cycles and large amounts of plutonium are present in spent fuel from civil nuclear power plants. The subject has been studied and reported on by the American Nuclear Society and a number of other studies (American Nuclear Society, 1996).

In the short-term weapons grade plutonium from the weapons production programmes is a significant proliferation risk. Weapons grade plutonium has 90% or more plutonium-239 which is the more suitable isotope for explosive applications. An important objective to ensure non-proliferation is to convert such plutonium to a different form, e.g. to the spent fuel standard (American Nuclear Society, 1996). However, the timescales for such action are relatively long, anticipated being as much as 15 years.

The longer-term issue is concerned with the increasing quantities of the plutonium being produced by the civil nuclear power programme. If it is sufficient to leave plutonium in spent fuel, how difficult is it for plutonium to be recovered, etc? There are issues associated with the choice of future nuclear power plants, e.g. whether a fast reactor will be built with a requirement for plutonium fuel. It will also depend on the utilisation or otherwise of advanced fuel cycles, e.g. MOX fuel. Either scenario would require the

Table 2.15. Reasons for deferment of plutonium in the civil nuclear fuel cycle

Minimise global nuclear catastrophe risk from irresponsible factions, e.g. terrorists
Limit military applications from civil programmes
Reduce the risk from adverse political changes in nations with existing arsenals
Remove international barriers to weapons stock destruction
It is not possible to guarantee protection despite IAEA safeguards and international monitoring
Separation of plutonium is not justified by current or anticipated market conditions for the next few decades

Cochran (1996).

separation of plutonium from spent fuel with the increased risks of such action on proliferation.

The threats of proliferation have been categorised in American Nuclear Society (1996) in terms of 'national' and 'sub-national' threats. National proliferation is defined as the use of weapons grade material or material separated in the fuel cycle for weapons devices, authorised by national government approval. Sub-national proliferation would relate to the threat of seizure of nuclear material by smaller groups of people, acting without the support of government.

Retaining plutonium in spent fuel is likely to be an effective deterrent against sub-national threats, but reprocessing, albeit at small scale, is within the capability of many industrialised countries and so the spent fuel standard barrier does not provide protection at a national level. It was mentioned in the previous chapter that IAEA have defined controls to provide a high degree of protection across many eventualities. Nevertheless, it is clearly the effectiveness (or lack of it) of the implementation of these controls that is the issue. It is an issue of increasing importance as the stocks of spent fuel continue to increase and as the radioactivity of older stocks of spent fuel diminishes, the recovery of plutonium becomes easier.

It is argued by Cochran (1996) that due to the general increase in terrorism in the world and for other reasons, there are strong reasons for deferment of the further chemical separation of plutonium at the present time (Table 2.15).

REFERENCES

American Nuclear Society (1996) Protection and management of plutonium. *Nucl. Energy*, **35** (4), 203–210.

Blee, D. (2001) Lessons learned? Selected public acceptance case studies since three mile island. *Nucl. Energy*, **40** (1), 33–35.

Birol, F. (2000) Nuclear power in the world energy outlook. *Nucl. Energy*, **39** (1), 49–54.

Blix, H. (1998) Energy and global sustainable development. *Nucl. Energy*, **37** (2), 113–118.

Bolton, C. (1996) Plant life extension for early nuclear power plants. *Nucl. Energy*, **35** (1), 53–57.

Chamberlain, L.N. (1997) Power for the next century. *Nucl. Energy*, **36** (2), 89–93.
Chung, K. (1998) The prospects and challenges for nuclear power in Asia. *Nucl. Energy*, **37** (3), 193–203.
Cochran, T.B. (1996) Plutonium: the international scene. *Nucl. Energy*, **35** (4), 211–222.
Digest of UK Energy Statistics, DTI, 2002.
DTI Energy White Paper (2003) Our energy future – creating a low carbon economy, TSO (The Stationery Office), Crown Copyright.
Duncan, I.J. (2003) What to do with nuclear waste. *Nucl. Energy*, **42** (3), 145–148.
Energy Visions 2030 for Finland (2003) VTT Energy, Edita Prima Ltd, Helsinki.
Fisk, D.J. (1999) The environmental impact of nuclear power. *Nucl. Energy*, **38** (2), 123–127.
IAEA-TECDOC-1084 (1999) Review of Selected Cost Drivers for Decisions on Continued Operation of Older Nuclear Reactors, May 1999.
IAEA Technology Annual Report (2002) IAEA web site http://www.iaea.org.
May, P. (2003) Supply chain approach & opportunities within Magnox, Twenty Fifth Meeting of the NIA Decommissioning & Radwaste Working Group, Warrington, 15th October 2003.
Mayson, R. (2003) A nuclear perspective, future power: meeting the challenges for electricity generation, BNES Seminar, Culham Science Centre, UK, 30 April 2003.
Mortin, S.J. (2000) Long-term challenges and continued operation of nuclear plant. *Nucl. Energy*, **39** (3), 171–174.
Mounfield, P.R. (1991) *World Nuclear Power*, Routledge, London.
Performance and Innovation Unit – February (2002), Cabinet Office, Crown Copyright.
Presentation to the BNIF/BNES Conference – Energy Choices (2002) The future availability of graduate skills, 5th December 2002.
Rougeau, J.P. (1997) Experience and prospects in reprocessing. *Nucl. Energy*, **36** (2), 123–125.
Ryhanen, V. (1996) Nuclear waste disposal – how to meet the concerns of the public. *Nucl. Energy*, **35** (5), 349–351.
Smitton, C. (1999) *Nucl. Energy*, **38** (2), 123–127.
Smitton, C. (2000) Magnox history, current status and future strategy. *Nucl. Energy*, **39** (3), 171–174.
Storey, P.D. (2001) Availability of skills and research capability. *Nucl. Energy*, **40** (4), 237–240.
Suzuki, A. (2000) Causes of the JCO criticality accident and lessons learned. *Nucl. Energy*, **39** (6), 373–380.
Twidale, D.W. (1999) Life extension of magnox nuclear power stations. *Nucl. Energy*, **38** (2), 107–112.
Wilmer, P. & Bertel, E. (2000) Nuclear power: a competitive option? *Nucl. Energy*, **39** (1), 55–60.
World Energy Outlook (1998) International Energy Agency, OECD, Paris.

Chapter 3
Operational Safety

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Chapter 3

Operational Safety

3.1. INTRODUCTION/OBJECTIVES

This chapter addresses the issues of operational safety for existing nuclear power plants. It concerns safety throughout all aspects of power plant operation and a number of topics are covered. There are diverse issues, some of which have already been introduced in the previous chapter. The management of radioactive waste is a particular concern and must be resolved, although there is positive progress on this issue in some countries. It is now recognised that human factor considerations can play an important role in maintaining plant safety. Increasing attention is being paid to improved operator training and other means of reducing the risk of human error. Another goal is to achieve closer collaboration between regulators and utilities. The regulation of an increasing number of privatised utilities is a present day issue. Another topic included is how the experience of many reactor years of operation can be utilised to improve future safety.

Operational safety is of paramount importance for the continuation of nuclear power, both nationally and internationally. The Chernobyl accident demonstrated all too clearly that the consequences of a major accident cannot be confined within national boundaries. Further, were such an accident to occur in the future, the nuclear industry would be unlikely to survive in most countries. Regarding continuous improvement, additional safety systems have been back-fitted to some of the older operating reactors. Extended accident management procedures have also been developed. Both of these are intended to extend the plant safety envelope.

3.2. INTERNATIONAL SUPPORT

Nuclear safety is a major topic of interest within the activities of a number of international organisations, see for example Hall (1998). The IAEA has taken a lead role in promoting the role of atomic energy in all aspects. This includes operational safety and the agency is helping to set up regulatory frameworks throughout the world. The IAEA is helping to define common standards and understanding, one example is the setting of agreed event scales, which aim to provide a safety significance marking for a particular event. Another example is 'safety culture', recently reported on by the IAEA's International Nuclear Safety Advisory Group. The intention is to instil an awareness of safety significance in all those responsible for the safety of a nuclear plant.

The IAEA carries out operational safety reviews through operational safety review teams (OSARTs). The OSARTs investigate particular operational safety issues, identify lessons learned and monitor corrective actions.

Another important international body is the World Association of Nuclear Operators (WANO), formed in 1989. The members of this organisation are solely utilities. WANO's aim is to maximise the safety and reliability of nuclear power plant operation through exchange of information amongst its members. WANO conducts many different programmes. These include exchange of operators between different stations. Operator exchanges have taken place between many Western and Eastern European countries. WANO has set indicators for plant performance (Table 3.1). Examples of such plant performance indicators relevant to safety include unplanned scrams, levels of radiation exposure, accident frequency and so on. WANO has also instigated various peer review programmes.

Prior to WANO, the Institute of Nuclear Power Operators (INPO) was set up by US utilities to promote safety culture in the US, working with other US organisations, the Electric Power Research Institute (EPRI) and the American Nuclear Society (ANS).

There have been a number of international initiatives to transfer Western safety culture to Central and Eastern Europe and Russia. Aid has been provided by the G7 countries via the European Bank for Reconstruction and Development (EPRD) and also by the European Commission through the Poland and Hungary Aid for the Restructuring of the Economy (PHARE) and the Technical Assistance to Commonwealth and Independent States (TACIS) programmes. Although much of this support has been spent on technical consultancy, a significant proportion has been spent on plant improvements and training of operator personnel.

Other activities either directly or indirectly supporting improved operator safety have resulted from the activities of the NEA of the OECD (NEA/CSNI/R, 2001).

Table 3.1. WANO performance and safety indicators

Unit capability factor – % of maximum energy generation that a plant is capability of supplying to the electrical grid
Unplanned capability loss factor – % of maximum energy that a plant is not capability of supplying to the grid because of unplanned energy losses
Unplanned automatic scrams per 7000 h critical – mean scram (automatic reactor shutdown) rate per year (approximately)
Collective radiation exposure – monitor of effectiveness of total personnel radiation exposure controls
Industrial safety accident rate – number of accidents resulting in lost work, restricted work or fatalities per 200,000 work-hours
Safety system performance – availability of three important standby safety systems at each plant
Fuel reliability – progress in preventing defects in the metal cladding that surrounds fuel
Chemistry performance – progress in controlling chemical parameters to retard deterioration of key plant materials and components during the operational lifetime

Carle (1997) and NEA/CSNI/R (2001).

Collectively these international programmes contribute to improved operational safety of the world's power plants. There are many areas of complementary and collaborative activities. The incident reporting system (IRS) for example is managed by both the IAEA and the NEA/OECD, and both liase with WANO. WANO collaborates with IAEA in many of its work programmes including the scheduling of peer reviews, operational safety review missions and so on.

3.3. SAFETY PERFORMANCE

Much experience has been gained from around 50 years of successful and largely safe operation of nuclear power plant around the world. Undoubtedly the operational safety of most reactor systems has been improved by the collective knowledge acquired from all types of reactors. Many of the principles for safe operation relate to plant management and other generic factors and are not specific to a particular type of plant.

In the first instance, safety must be built into the plant design. This is usually referred to as engineered safety. Good design can prevent significant accidents through the intervention of good safety systems. Conversely there are examples where less good design has resulted in very significant major accidents. New designs will benefit from previous operating experience.

Operational safety has generally come to relate to the performance of plant personnel and the management of plant safety at the plant. The performance of management and staff can be judged against a number of performance indicators. Recent WANO data for collective radiation exposure and industrial accident rate are shown in Figures 3.1 and 3.2, respectively. These show steadily improving trends.

Although there may be differences in detail, performance indicators utilised by different utilities have much in common. For example, in the UK, BNFL/Magnox Generation, in a recently published review of station performance, consider indicators

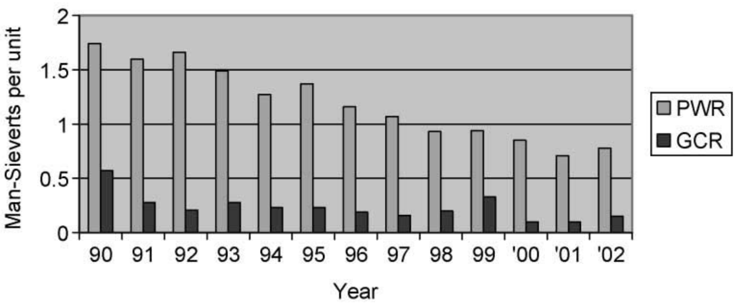

Figure 3.1. Collective radiation exposure for PWRs and GCRs (WANO). Source: WANO (2002).

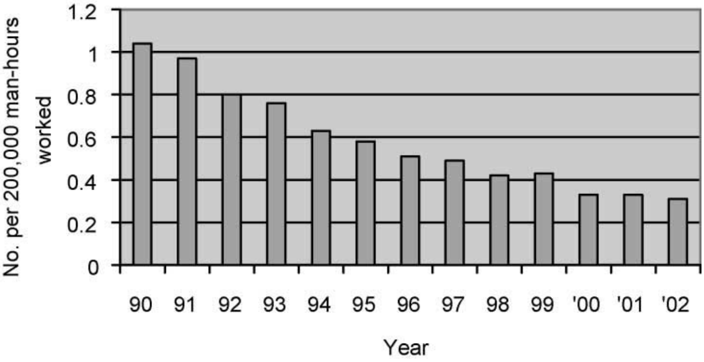

Figure 3.2. Industrial safety accident rate (WANO). Source: WANO (2002).

such as collective radiation dose, lost workday rates, the number and severity of events and the number of automatic trips (Marchese, 2000). Selective data are shown in Table 3.2 and these also indicate an improving trend.

There are a number of international standards which industry can use in assessing and improving performance. The internal safety system (ISRS) of Det Norske Veritas consists of different elements of performance. For example, element 1 relates to 'leadership' in putting emphasis on safety and reliability in achieving high standards of performance. The plant in question is then rated at a particular level. The system was used by BNFL/Magnox Generation in the review referred to above.

WANO performance objectives and criteria also include management performance. There is an increasing realisation that improvement in company business performance is commensurate with improvement of safety. A company needs safe reliable operation in order to be competitive. As inferred above then, both business and safety performance depend heavily on plant personnel.

The European for Quality Management (EFQM) model has been developed to provide a method for reviewing how management processes are actually working in practice. One of the features of EFQM is that both business and safety objectives and standards

Table 3.2. Safety improvements in Magnox plant

Measure	End 1980s	End 1990s
Collective dose (man Sv/reactor)	~0.5	~0.2
Lost work day case rate (per 10^5 h)	~0.9	~0.3
International nuclear event scale (INES) 1 (annual total)	~35	~14
INES 2 (annual total)	~2	0

Data from Marchese (2000).

should be implemented at all levels through the company. Regarding personnel development, the UK Investors in People (IIP) standard has been adopted by many companies in promoting the well being and development of their staff.

3.4. RADIATION PROTECTION

3.4.1 Individual Protection

There are approximately 11 million radiation workers worldwide, (IAEA/NSR/2002, 2003). Standards of radiological protection in the nuclear industry are very high and are probably more advanced than those in practice in the non-nuclear industry. The science is also very mature although there is some need for greater harmonisation of terminology, quantities and units. The IAEA works closely with employers, regulators and workers through the International Labour Organisation (ILO) in the continuing development of safety standards for occupational protection. The agency is also involved with radiological protection of individuals in general, including for example the protection of patients undergoing radiology.

3.4.2 Discharge Monitoring

The release of radioactive materials from nuclear power plants is routinely monitored. Under normal operating conditions such releases are very small and are difficult to measure against background levels, even with modern instruments.

Since the early days of nuclear power the routes for radioactive pathways to man have become much better understood. The sensitivity of instruments and measurement techniques has markedly improved. The same is true for the techniques for analysing results.

With improved monitoring techniques have come more rigorous monitoring standards imposed by nuclear regulators. In the early days, the available instruments were in some cases, not able to distinguish station releases from the background and standards were less rigorous, reflecting this limitation. For example, the Federal Standard in the US was 5 mGy per year (Mounfield, 1991) up until 1971. After this time, more stringent limits were introduced at site boundaries; the limits were set down to 0.1 mGy per year for external gamma radiation from nuclear effluents.

For monitoring higher levels, passive devices are used, e.g. thermoluminescent or film dosimeters. For greater sensitivity, high-pressure ionisation chambers are available for measuring gamma-emitting radionuclides such as Iodine-131. Gamma ray spectroscopy using multichannel analysers provides a capability to analyse large numbers of environmental samples.

Routes for release that are monitored include, e.g. ventilation stacks, routes for discharging wastewater through cooling waters to the sea, rivers, etc. Data are supplied to national radiological protection bodies and also international bodies, e.g. United Nations Committee on the Effects of Atomic Radiation, UNSCEAR (U.S. Regulatory Commission, 1980).

Releases to the environment during normal operation are now set to very low limits. For example, Swedish Regulations (U.S. Regulatory Commission, 1980) require that any releases to the environment during normal operation must result in a dose equivalent of less than 0.1 milli-Sievert (mSv) per year to nearby residents. Releases to the environment are a subject of continuing interest to the IAEA (IAEA/NSR/2002, 2003). The IAEA is promoting an international conference on the protection of the environment from the effects of ionising radiation to be held in Sweden in 2003 (International Conference, 2003). This will aim to achieve an international consensus to form the basis of the agency safety standards in this field.

In addition to offsite releases, onsite releases and exposures are also monitored. For workers in the nuclear industry, there has been a general reduction in operational exposures since the start of nuclear power plant operation. In the UK (Hughes, 1996) and elsewhere, this has been driven by changes in regulatory requirements including the as low as reasonably achievable (ALARA) principle and more restrictive dose limits. It has also been driven by the support of industry and the availability of improved radiation protection methods. Table 3.3 shows the reduction in UK worker dose and discharges during the early 1990s. Figure 3.3 shows the reducing trend of maximum individual dose at the UK Aldermaston establishment during the 1990s.

The 1990 International Commission on Radiological Protection (ICRP) recommended the main limit as an annual average of 20 mSv over a five-year period, not exceeding 50 mSv in any single year. Annual dose limits are currently 20 mSv per year with a lifetime dose of a few hundred mSvs. In the UK, now only a handful of workers receive doses over 15 mSv per year (Hughes, 1996). In addition to individual dose, measures of

Table 3.3. Worker dose and discharges

Dose measure	1990	1996
UK classified workers receiving more than 10 mSv (excluding miners) (HSE, 1998)	~1500	~100
UK annual collective Dose for classified persons (excluding miners) receiving more than 0.1 mSv (man Sieverts) (HSE, 1998)	~80	~35
Typical annual UK individual dose from civil discharges (μSv)	~35	~1
Collective UK dose truncated to 500 years from UK civil discharges (mSv) (Bexon, 2000)	~128	18

Data from Peckover (2002).

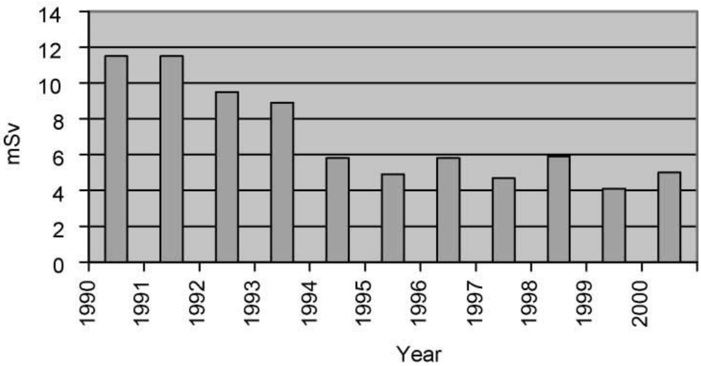

Figure 3.3. Maximum individual dose at AWE Aldermaston. Source: Sallit (2002).

occupational exposure at a particular power station are also used, by considering the total dose of all personnel who have received a measurable exposure.

3.5. WASTE MANAGEMENT SAFETY

The main objective in the management of radioactive waste is to ensure the protection of the public, workers and environment by isolating all hazardous material from the biosphere. Technologies are required for the various stages of waste management, i.e. the handling, temporary storage and long-term disposal. To ensure safety, the handling, storage and disposal of all waste materials arising from all plant operations is carefully managed according to the degree of hazard.

Wastes are obtained from all stages of the fuel cycle, from uranium extraction, refining, reactor operation and decommissioning. Waste can arise in gaseous, liquid or solid forms. Some of these waste products are radioactive with varying degrees of activity. They include low-level radioactive spoil from uranium mining residues and uranium and plutonium residues from fuel fabrication.

Reprocessing plant operations produce waste of medium level of radioactivity, arising from various waste streams, and hulls from residual cladding and support materials from fuel elements. There are other low-level wastes from reprocessing operations, including miscellaneous items such as gloves, containers, etc.

The reactors produce highly radioactive waste in the form of spent fuel; this issue is discussed further in the next section. They also produce inert gas (waste) from fractured fuel elements, liquid waste in the form of tritiated water and solid waste, e.g. filters and resins. The latter arise from water treatment plants; these resins are used to clean up primary system fission or corrosion products.

Decommissioning also produces mildly radioactive structural materials.

Drainage water from reactor support systems, fluids from decontamination operations and ion-exchange resins in the form of liquid effluents are collected in tanks. These effluents are then categorised and distributed to various sub-systems depending on their activity or impurity content.

Low-level effluents are discharged under controlled conditions back into the reactor plant. Solid low-level wastes may be compacted and then encased in stainless steel drums. Intermediate effluents may be treated with ion-exchange filters or evaporated, with the purified water returned to the reactor coolant system. Active resins or concentrated active liquids are stored in tanks, to allow for decay of the shorter lived isotopes, before being sent for waste treatment. Solid wastes such as filter resins evaporation residues are treated and then cast in concrete.

More details on the scale of the global nuclear waste management issue and present and possibly future containment practices are given in Chapter 6.

3.6. SPENT FUEL

The primary objective of present day fuel cycles is to optimise energy production, while at the same time minimising costs and maximising safety. This has given rise to a number of fuel cycle variants to be considered in which fuel is recycled between different reactor systems, see for example, Ion and Bonser (1997). This is discussed more fully in Chapter 5. There are particular operational safety issues relating to the management of spent fuel, etc., as mentioned in the previous section. These are concerned with safe storage practices, maintenance of sub-criticality in fuel storage ponds and flask transport safety and security.

Spent fuel represents the most highly radioactive waste arising from the fuel cycle, due to the presence of particular fission products. It, therefore, represents the significant waste radiological hazard associated with nuclear plant operations. It may also include fissile and fertile uranium or plutonium and possible breeding products, which might be re-utilised by recycling or reprocessing. The fission product radiological hazard will remain, however, whether or not these materials are recovered.

Policy issues are likely to play a continuing role in the development of advanced fuel cycles. Reprocessing and recycling, together with improved uranium resource management can lead to a reduction in waste volumes and toxicity, thus improving the sustainability of nuclear power plant operation but this may not be a preferred option due to economic reasons or proliferation concerns. Recycling of plutonium in LWR MOX cores reduces the spent fuel radiotoxicity by a factor of 3 if the MOX fuel itself is not recycled (Bertel and Wilmer, 2003). Multiple reprocessing and recycling can reduce radiotoxicity further.

Reprocessing practices also reduce the volumes of radioactive waste significantly. Each tonne of spent fuel contains about 1.5 m^3 of high level waste. After reprocessing, less than 0.5 m^3 of waste remains, including 0.115 m^3 of vitrified high level waste and 0.35 m^3 of intermediate waste. Additionally further compacting can be carried out after disposal.

3.7. OPERATOR TRAINING

The accident at TMI-2 set in motion action plans for improved operator training in many countries (U.S. Regulatory Commission, 1980). A particular consequence was to recognise that training must be broadened to include more emphasis on operational incidents and accident circumstances. Further training should not just be for operators but training of maintenance and all personnel connected with the plant operation should also be included. Recommended practices are given in Table 3.4.

Training needs to be provided at various levels. It must include new staff but also employees who might be changing jobs through transfer, redeployment or promotion (Leclercq, 1986). It should occupy a reasonable amount of time; e.g. French nuclear power plant workers receive typically 80–90 h training per year.

Present day training includes extensive use of simulators and electronic-aided teaching procedures. These are used to teach the candidates the single component aspects of plant operation, e.g. the chemical and volume control system, control of the reactor and managing the balance of plant function (e.g. the turbine-generator performance).

Table 3.4. Personnel training

Good plant practices and personnel development	Description
Training	Continuous training philosophy, learning and sharing knowledge
Examination	On-the-job and examinable training
Simulators	Specific operator training including simulators and other relevant plant operations factors
Quality	Training quality improvement taking advantage of different tools available
Improvement of human factors training	Training materials for plant personnel working in multiple areas
Expansion of training	More in-depth training of radiation workers to improve contamination control
Performance improvement	Individual training records to be kept by each operator
Management	Accountability of line management with support from training organisations

IAEA-TECDOC-1098 (1999).

Full scope simulators provide an exact replica of the control room and enable a complete simulation of plant behaviour. The candidate can, therefore, be subjected to all forms of plant condition from start-up, shutdown and response to faults.

The training of maintenance staff is equally as important as the training of operators. Their training involves acquisition of the necessary technical expertise but also they need to learn the organisational and communication skills to manage a large workforce. They must acquire the necessary skills to work and be compliant with the various safety instructions and other working conditions, including working under the pressure of meeting challenging delivery deadlines.

The training of power plant personnel has been addressed by the IAEA. A guidebook was published which was concerned with the training to establish and maintain qualified and competent operations' personnel for plant operation. The guidebook has been subsequently updated in 1996 (IAEA, 1996) and the latest version includes worldwide experience that has been gained since earlier publication.

IAEA recommendations are that there should be a systematic approach to training for nuclear power plant personnel. The approach should not only cover the operators but should cover the role and responsibilities of management including the training of management. The approach should include the evaluation of training methods and look for ways of making training more effective. It should also cover the organisations involved in providing the training.

3.8. HUMAN FACTORS

Human factor issues affect most aspects of plant design, operation and maintenance. The subject has received increasing attention over recent years from both regulators and utilities. Operating experience has shown that plant personnel and the systems within which they operate, play a very important contribution to safety.

There have been recent efforts in human factors engineering, ergonomics, and biomechanics to improve understanding and safety operation (Ramsey, 1998). These have also included human/machine interfaces and the development of special purpose systems. These must manage various data inputs relating to information gathering and output to maximise human and machine performance. Techniques for human error rate prediction (THERP) have been established. These compile human error rates for various industrial tasks. Comparisons with other industries indicate that nuclear facilities generally meet very high comparative degrees of safety. Studies of human attitudes and physical limitations are given in USNRC (1992). An account of workers responses to events and their compliance with control measures is given in SOER92-1 (1992).

Human factors' issues have been recognised by the UK regulator via specific safety assessment principles addressing human factors issues (Dixon, 1998).

One method of improving safety is to identify factors that impact on performance of a particular job. Utilities have developed a number of techniques to help them to analyse particular tasks. Factors that impact on performance include design of interfaces, the procedures in place and staffing levels and training. Good practice guidelines that have been recommended include the adoption of user friendly operating instructions and peer review of proposed changes, etc.

The HSE in the UK has also recognised the importance of broad-ranging organisational factors including safety management systems and safety culture (Dixon, 1998) (Table 3.5). The establishment of good safety culture within organisations is important to ensure the implementation of safety principles at all levels within the plant. The HSE, in common with most other safety authorities, believes that the licensee should own its safety cases. This is particularly important today in many countries including the UK, where nuclear industries are undergoing rapid change and where there is increasing use of contractors.

BNFL/Magnox Generation commissioned a study to establish the relationships, if any, between employee safety awareness and safety performance (Spooner and Vassie, 1999). This study identified five factors – training/experience, safety initiatives, communication, organisation and personnel.

The participants in the study felt that safety awareness was developed partly via common sense and partly via specific skill training. Personal experience, particularly of an unsafe event, was not surprisingly, highly influential on an individual's safety awareness, but in addition learning from a colleague's experience was also influential. Training/experience were considered to be key influences in employee safety awareness.

Safety initiatives such as 'near miss' reporting were regarded as effective in promoting safety awareness. It found that performance-related bonus schemes, including the achievement of specific safety targets did not significantly influence safety-related behaviour. This would appear to contrast the situation in certain organisations in North America.

The study appeared to show that passive forms of communication, e.g. notices, had less impact than verbal communication; e.g. team briefings were found to be more effective. The employment of active communications' systems, e.g. PA and VDU systems, was also considered to be more effective.

It was concluded that improved communication of learning events was useful and that the organisation should be in place to do this. How the organisation responded in resolving

Table 3.5. Human factors issues

Impinge on plant design, operation and maintenance
Task analysis – procedures, training, interface design, staffing levels
Safety culture at all levels – organisation, plant management, staff
Licensee ownership of safety case – cf. use of contractors

Dixon (1998).

safety issues was also considered but the participants did not feel that this contributed significantly to safety awareness. However, further work was required.

Personal relationships, including interactions within a group and responsibility for others were considered to influence safety awareness significantly.

Although the study cited was specifically for the BNFL/Magnox Generating Group, it was felt that the conclusions have a wider application to other similar organisations.

Safety management and safety culture have been reviewed in a recent IAEA international conference on safety culture in nuclear installations held in Rio de Janeiro in December 2002 (IAEA International Conference, 2002; IAEA/NSR/2002, 2003). This conference confirmed that safety culture is now regarded internationally as an important element of nuclear safety. The IAEA Nuclear Safety Standards Committee endorsed a proposal in 2002 to develop safety standards specifically addressing safety management and culture. Two particular issues identified at this conference were that although safety culture is now embraced by top management, there is still a need to broaden appreciation through to the shop floor. It was also noted that safety culture is being embraced more enthusiastically in countries with a developing industry, than in those which had long-established nuclear programmes.

3.9. REGULATION

The design, construction and operation of nuclear power plants are carried out against agreed safety principles that are set by national safety authorities. Activities are regulated by legislation and compliance with safety principles is enforced by national regulatory authorities. The primary objective is to ensure the risk of harm to the general public is acceptably low.

There are differences in scope and emphasis across different regulatory authorities. In the US, the regulatory regime is very prescriptive. The US Nuclear Regulatory Authority (USNRC) defines detailed safety rules that have to be demonstrated by the utility before a licence can be granted. Regulatory guides are available to help a utility in demonstrating compliance. The USNRC is a large body and incorporates a standardised regulatory role.

The situation in other countries may be different and much less prescriptive, as is for example the case in UK. In the UK, the NII issues safety principles that have to be satisfied, but do not require any specific methodology. It is up to the utility to define an appropriate methodology. He must then convince the safety authority that this is adequate for ensuring that the safety principles are met.

The plant licensing procedures certainly within the Western world (and increasingly within most safety authorities worldwide) follow along similar lines, see for example, Pershagen (1989). The requirement for detailed safety analysis reports (SARs) at different stages in the plant licensing cycle is one example (Table 3.6). Prior to construction, the

Table 3.6. Established licensing procedures

Preliminary Safety Analysis Report (PSAR)	Detailed description of preliminary plant design, performance and safety policy objectives
Final Safety Analysis Report (FSAR)	Usually encompasses the PSAR and includes how the plant complies with safety requirements
Periodic Safety Reviews (PSR)	Systematic regular re-evaluation of the overall plant safety, typically every 10 years

U.S. Regulatory Commission (1980).

applicant produces a preliminary safety analysis report (PSAR). The PSAR contains a detailed description of the site and all aspects of the plant design and performance. This would include a description of the engineered safety features (ESFs) and the analysis of design basis accidents. During the construction the final safety analysis report (FSAR) is produced which gives a detailed description on how the plant will be operated to meet the safety requirements.

Finally commissioning tests are carried out, both pre-criticality to check system performance and then after fuel loading at low power, before permission to go to full power is granted. During normal operation, regular reports are provided to the safety authorities, including details of plant performance and safety-related monitoring measurements, radiation exposures, etc. In addition, any abnormal events would be reported. Another generally accepted practice is to carryout PSRs, as referred to in Chapter 2. These are usually required to take place at least within every 10 years. The PSR is a wide-ranging review to look at all issues of plant operation and safety taking into account operational history and the experiences from operation of similar plants worldwide.

3.10. UTILITY AND VENDORS

A commonly accepted principle is that the direct responsibility for safety of a nuclear plant rests with the utility. This contrasts the role of the regulator whose function is to set the safety goals and to ensure these are met. As for differences in regulatory focus across different countries, there are also differences in relationship between utility and regulator. This relationship is more formal in some countries than others. However, in most countries, there is a desire from both utility and regulator to promote an increasingly collaborative working relationship and encourage open dialogue between the two parties.

It is the responsibility of the licensee to have in place emergency operating procedures for the plant and also emergency planning procedures for the whole nuclear plant complex (Pershagen, 1989). These include instructions for plant operation for accidents within the design basis but also procedures for severe accidents beyond the design basis (Table 3.7).

Table 3.7. Objectives of operating rules and accident management

Operating rules for design basis accidents (event oriented)	Assure safety during operation Maintain the plant operation within the limits imposed by the design and safety specifications
Emergency operating procedures for beyond design basis severe accidents (symptom oriented)	Ensure sufficient sub-criticality Maintain adequate core cooling Minimise radioactive releases

Pershagen (1989).

Emergency planning procedures include the establishment of an organisation to implement the plan, including any required accident management actions. Emergency planning procedures beyond the plant boundary usually are the responsibility of other local and/or national authorities. However, the licensee must be prepared to liase and co-ordinate operations with these authorities.

The utilities are regularly audited by safety authorities to ensure that all the required frameworks/organisations are in place for the continued safe operation of their nuclear plants. An important principle is that the safety case is 'owned' by the plant, i.e. that a utility has in place a sufficient number of adequately trained staff who understand the relevant issues and are suitably qualified and experienced personnel (SQEP).

Reactor vendors clearly perform an integral part in ensuring the operational safety of a plant. They may be called upon by the utility to provide services not just at the design and initial licensing stages but also during the lifetime of the plant. There may be a requirement to back-fit more efficient safety systems, or to purchase fuel from a new fuel vendor (the latter is more performance- than safety-related). In the US, there has been a push by some vendors to license specific designs with the USNRC. It is expected that this would substantially simplify the licensing process of that particular design, e.g. in countries outside the US.

3.11. PERIODIC SAFETY REVIEWS

In many countries periodic safety reviews are required to be carried out by the plant operator as a condition for his site licence. The primary objective of most PSRs is to undertake a detailed and comprehensive review of the safety of the plant, taking into account operational safety, the possible deleterious effects of ageing, and also advances in safety standards since the original construction or time of the last review.

Periodic safety reviews are usually complementary to the normal regulatory reviews that are carried out, e.g. between fuel cycles and do not affect them. PSRs have developed for a number of reasons. Public confidence has diminished and regulatory requirements have become more stringent over the past decade or so driving a demand for higher standards of safety not only in new plants, but also in currently operating plants. Perhaps rather more importantly though, experience has shown that there are positive benefits from PSRs to both safety and performance and they are supported by both operators and regulators.

A list of safety issues identified for PSRs is given in Table 3.8 together with a framework for review that was endorsed by the 1991 IAEA Safety Conference (Goodison, 1997).

A review of experience of PSRs has been published in CEC Working Group (1990) and Goodison (1997). It concludes that PSR practices show considerable commonalties. This is particularly so within the EC due to similarities in regulatory regimes within the EC countries. At the top level, the procedure is broadly similar. There is agreement of the scope between the licensee and the regulator. The licensee undertakes the review, implements the modifications and reports to the regulator. The procedure is then followed by review by the regulator and the identification of any further modifications. Finally agreement is reached between the licensee and the regulator on how to fulfil the agreed programme.

The differences in PSR practices depend mainly on the methodology, the standards and scope that are adopted. These differences might relate to the standards for radiological protection or on the level of redundancy and diversity of the safety systems. The criteria for PSAs are also not universally agreed. There are also differences in the periodicity requirements for PSR reviews.

The potential benefits of PSAs include improved safety via the implementation of modifications to an improved safety level (closer to that of a modern plant), including the

Table 3.8. Safety issues to be addressed in PSRs

Safety issues	Recommended procedures for assessing each issue 1–5
Lessons learned from operational experience, locally, nationally and internationally	Assess each issue with current methods to determine the safety status Compare the safety status with current standards
Changes in safety standards and practices	Identify shortfalls Assess the safety significance of any shortfalls and carry out remedial measures
Equipment qualification and ageing effects	Implement practicable modifications and assess safety significance of remaining shortfalls
Safety culture On-site emergency arrangements	Repeat for each issue

CEC Working Group (1990) and Goodison (1997).

Table 3.9. Benefits of PSRs

Gain indication of safety level compared with modern plant and identify shortfalls from current safety standards and best practices
Improve plant routine operations including optimisation of maintenance, test and inspection techniques and improve plant availability
Identify strengths and weaknesses of personnel
Gain improved understanding of plant safe working life and life-limiting causes
Improve regulator confidence in the continued safe running of the plant, improve the licensee's confidence for future planning and investment and improve national public and international confidence

Goodison (1997).

identification of short-falls in present practices and improved confidence (in the regulator, operator, public, etc.) (Table 3.9).

The first comprehensive review of the plant is usually the most demanding. Subsequent reviews would be expected to be quicker (and cheaper). Initial PSRs, of very old plants may require extensive modification or possibly result in closure. For future plants, initial PSRs might be expected to be less onerous.

3.12. SAFETY IMPROVEMENTS

Within present day generation plants, there is a continuing requirement to improve safety and performance. There has been a major investment in experimental and theoretical programmes of work to support this objective. Component research has been carried out to back-fit safety systems on older reactors. Better understanding has led to the development of additional accident prevention and mitigation guidelines. It is also leading to proposals for new systems, e.g. to reduce releases in severe accidents.

There has been a decade of safety upgrades and improvements in nuclear power plants in the EU Accession countries. The main design safety issues are associated with re-licensing, plant life extension, ageing and periodic safety reviews. Another issue is the completion of nuclear power plants that have been left partially built for a number of years (IAEA/NSR/2002, 2003).

3.12.1 Back-Fitting Safety Systems

It has been recognised over many years that the 'defence-in-depth' principle is fundamental to the design of nuclear power reactors and other types of nuclear plant (Table 3.10). The important feature is the requirement that multiple barriers exist against the release of radioactivity to the environment. The defence-in-depth principle is generally assessed using either or both deterministic or probabilistic methods.

Various means of strengthening the defence-in-depth principle are being considered in current generation reactors and indeed implemented, with respect to accident prevention

Table 3.10. Defence-in-depth

Level	Measures	Systems/Principles
1	Preventative	Operation/Control systems
		Inherent design attributes
		Safety margins
		QA
2	Protective	Safety systems
		Redundancy
		Diversity
		Segregation
3	Mitigative	Containment
		Activity removal systems
		Remote siting
		Emergency preparedness

International Nuclear Safety Advisory Group (1988).

(Hogberg, 1998). Additional levels of protection have been installed in many European and other reactors worldwide. In particular measures have been taken in a number of countries to improve the capability of existing components to withstand severe accident loads. The main objective is to mitigate the release of radioactive isotopes to the environment, particularly iodine and caesium. These measures have been complemented with the development of severe accident strategy improvements.

Clearly there are economic and technical constraints on back-fitting improvements in existing reactors. There are many types of design in operation and the feasibility of such improvements is design specific. Nevertheless significant improvements have been achieved at acceptable cost. Many of the desired measures have been identified in Periodic Safety Reviews, which are now a common-place regulatory requirement in most countries. They are being introduced within modernisation programmes, which may also be in place for other reasons, e.g. to replace out-of-date systems or instrumentation that has become too costly to maintain. There may be a requirement to improve the older operating plants to a standard commensurate with later models. If this is not achieved, it may be necessary to shut the older plants down.

It has been realised for many years that the defence-in-depth in many of the earlier Russian designed reactors only applies to a much more limited design basis than Western reactors. The safety of VVER and RBMK reactors has been extensively studied in a number of international projects over the last decade. Numerous safety recommendations have been made, including back-fitting of safety systems, etc. Some of these plants are operating in the EU Enlargement Countries, which will be joining the EU over the next few years. There is, therefore, a driver to accelerate the safety improvement process.

Table 3.11. Examples of back-fits on current plants

Availability of additional water-delivery sources
Filtered venting
Hydrogen control with ignitors and catalytic recombiners
Improved safety valves
Reinforcement of containment penetrations

Sehgal.

A number of safety improvements have been recommended for the early VVER-440 designs in respect of control of the reactor pressure vessel embrittlement, improved emergency core cooling systems and, improved reliability of residual heat removal systems. Additionally there are recommendations for improved instrumentation and control systems, including the reactor protection and shutdown systems and improved capability of the confinement to limit radioactive releases. Safety improvement programmes are underway to address these concerns.

There were greater drives for immediate safety improvements of RBMK designed reactors in the wake of the Chernobyl accident. Some of the early plants have now been shutdown but a number of safety improvements have been implemented in the newer RBMK reactors still in operation.

For example the Ignalina power plant in Lithuania (an EU Candidate country) has recently undergone international peer review and various short-term safety improvements have been recommended. These relate to control and protection system reliability, the structural integrity of the major primary circuit components and the confinement function, improved emergency operating procedures, and the need to address fire hazards that could impact safety systems.

The large amount of severe accident phenomenological research carried out for Western water reactors has led to various mitigation measures being introduced and back-fits to be implemented (Table 3.11). The safety of current generation plants has been substantially improved by the development of this knowledge base. Containment research has been supported by experimental programmes on the removal of aerosols with sprays and on the modes of hydrogen combustion using igniters. The results of research for present day reactors are also benefiting the designs for future plants. Many of these plants include severe accident mitigation concepts in their design. Advanced designs include measures for improved in-vessel coolability of debris and ex-vessel debris coolability and retention. These are discussed in the subsequent chapters.

3.13. SEISMIC EVALUATION

Seismic evaluation or re-evaluation is an important issue with many existing nuclear power plants (IAEA/NSR/2002, 2003). This may be required to take account of new information that has come forward since the original evaluation. In some cases better margins can be demonstrated with the availability of new analysis techniques. It may be that older conservative safety margins are not considered sufficient in the light of present day requirements or that original evaluations were inadequate.

There are a number of supporting facilities that also require seismic evaluation. These include laboratories, research reactors and fuel cycle facilities. In general the cases for these facilities are less advanced and they present a wide range of different situations. Seismic evaluation of existing nuclear facilities is the subject of a recent IAEA international meeting in 2003 (IAEA/NSR/2002, 2003; IAEA International Symposium, 2003).

A number of IAEA Member States have on-going seismic upgrading programmes to improve the safety of their operating plant.

3.14. RESEARCH REACTORS

There are several safety issues associated with the operation of research reactors that are being considered by the IAEA (IAEA/NSR/2002, 2003). There exist reactors that have been shut down for long periods with no definite plans concerning their future, i.e. whether and how they might be restarted or decommissioned. Most of the reactors within IAEA Member States are in countries where there is an established regulatory regime that covers all nuclear installations.

Another issue that has been identified has been the storage of spent fuel and nuclear waste at research reactor sites. An IAEA international conference on research reactor utilisation, safety, decommissioning and fuel and waste management will take place in 2003. This will provide a forum for all interested parties, reactor operators, vendors and regulators to share experiences and develop priorities (IAEA International Conference, 2003a).

3.15. SECURITY

The security of nuclear installations has become an issue following the concerns of global terrorism in recent years. Draft guidelines are being drawn up by the IAEA to enable utilities to carry out self-assessment of the safety and security of their installations. There is a general impetus to promote interaction between staff from safety and security backgrounds and to harmonise terminology. The safety and security of radioactive sources has been identified as an issue by IAEA, following the accidental overexposure of individuals from ‘orphan’ sources and from concerns arising out of September 11, 2001.

An IAEA international conference on the security of radioactive sources addressed this issue in 2003 (IAEA International Conference, 2003b).

REFERENCES

Bertel, E. & Wilmer, P. (2003) Whither the nuclear fuel cycle. *Nucl. Energy*, **42** (3), 149–156.

Bexon, A.P. (2000) Radiological impact of routine discharges from the UK civil nuclear sites in the mid-1990s. NRPB, Report NRPB-R312.

Carle, R. (1997) WANO: aims and achievements. *Nucl. Energy*, **36** (4), 265–267.

CEC Working Group on the safety of reactors, periodic safety re-evaluation in NPPs in EC Member States, Finland and Sweden, EUR 13056, 1990.

Dixon, E.M. (1998) Experience with the engineering and operational SAPs. *Nucl. Energy*, **37** (1), 33–37.

Goodison, D. (1997) Periodic safety reviews of nuclear power plants. *Nucl. Energy*, **36** (2), 137–142.

Hall, R.W. (1998) Safe nuclear operations – a world perspective. *Nucl. Energy*, **37** (2), 83–87.

HSE (1998) Occupational exposure to ionising radiation 1990–1996: analysis of doses reported to the Health and Safety Executive's Central Index of Dose information.

Hogberg, L. (1998) Reactor safety, European Commission, European Seminar, Brussels, vol. 1, XII/0318/98.EN, 14–15 October 1998.

Hughes, J.S. (1996) Trends in occupational exposure within the UK civil nuclear industry. *Nucl. Energy*, **35** (4), 249–252.

IAEA (1996) *Nuclear Power Plant Personnel Training and Its Evaluation: A Guidebook*, Technical Report Series No. 380, IAEA, Vienna, ISBN 92 0 1014961.

IAEA International Conference on Safety Culture in Nuclear Installations held in Rio de Janeiro, December 2002.

IAEA International Conference on Research Reactor Utilisation, Safety, Decommissioning and Fuel and Waste Management, Santiago, 10–14 November, 2003a.

IAEA International Conference on the Security of Radioactive Sources, Vienna, 10–13 March, 2003b.

IAEA International Symposium on Seismic Evaluation of Existing Nuclear Facilities, Vienna, 25–29 August 2003.

IAEA-TECDOC-1098 (1999) Evaluating and improving nuclear power plant operating performance, July 1999.

IAEA/NSR/2002 (2003) *Nuclear Safety Review for the Year 2002*, International Atomic Energy Agency, August 2003.

International Conference on the Protection of the Environment from the Effects of Ionising Radiation, Stockholm, 6–10 October 2003.

International Nuclear Safety Advisory Group (1988) *Basic Safety Principles for Nuclear Power Plants, a Report*, Safety Series No.75-INSAG-3, International Atomic Energy Agency, Vienna.

Ion, S.E. & Bonser, D.R. (1997) Fuel cycles of the future. *Nucl. Energy*, **36** (2), 127–130.

Leclercq, L. (1986) The Nuclear Age, Sodel.

Marchese, C.J. (2000) Improvements in operational safety performance of the Magnox power stations. *Nucl. Energy*, **39** (5), 279–283.

Mounfield, P.R. (1991) *World Nuclear Power*, Routledge, London.

NEA/CSNI/R (2001) Summary Report on the Use of Plant Safety Performance Indicators, 11, September 2001.

Peckover, R.S. (2002) Radiation protection at low doses – benefits and challenges. *Nucl. Energy*, **41** (5), 333–338.

Pershagen, B. (1989) *Light Water Reactor*, Pergamon Press, New York.

Ramsey, C.B. & Modarres, M. (1998) *Commercial Nuclear Power: Assuring Safety for the Future*, Wiley, New York.

Sallit, G.C.R. (2002) A decade of dose reduction at AWE Aldermaston: what worked and what didn't. *Nucl. Energy*, **41** (4), 265–269.

Sehgal, B.R. (2003) Assessment of severe accident research phenomenological studies in laboratories and mitigation measures in reactors, EU Research in Reactor Safety, FISA, Pre-proceedings 10–13 November 2003.

SOER92-1 (1992) *Reducing the Occurrence of Plant Events Through Improved Human Performance*, Institute of Nuclear Power Operations, Atlanta, GA.

Spooner, K. & Vassie, L. (1999) The influence of workplace factors on employee safety awareness in the nuclear industry. *Nucl. Energy*, **38** (2), 91–97.

U.S. Regulatory Commission (1980) NRC Action Plan developed as a Result of the TMI-2 Accident, USNRC Report NUREG-0660.

USNRC (1992) Human performance in operational events, reactor operational analysis branch, office of analysis and evaluation of operational data, December, 1992.

WANO 2002 Performance Indicators, http://www.wano.org.uk

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Chapter 4

Operational Efficiency

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Chapter 4

Operational Efficiency

4.1. INTRODUCTION/OBJECTIVES

The needs for operational efficiency and reliable performance are clearly important to the continued operation of current generation plants. This is particularly so if the operating utility has to compete in an open electricity market. This chapter concerns the issues of power plant operation in relation to performance. In general, a significant factor in achieving maximum plant performance is to operate as close to the operating margins (within the constraints of safety limits) as possible. Another factor is to ensure the plant is on load for optimal periods, i.e. plant trips, maintenance and refuelling outages are minimised. These items are covered in detail in this chapter.

Optimised fuels are being developed in order to generate more power within the operating margins, to deliver improved fuel performance, and to extend the duration of individual fuel cycles. These fuels are being developed for both currently operating and evolutionary plants. Nuclear fuel cycles are covered in Chapter 5; a whole chapter is devoted to these topics in view of their importance to operational matters.

Deregulation of the electricity industries in many countries is an important driver for utilities to improve operational efficiency and performance and reduce costs. This is likely to be an increasing global trend in the future. Deregulation took place in the UK some years ago following the break-up of the Central Electricity Generating Board. Deregulation of the US electricity industry has occurred over the past decade, following the 1992 US Energy Policy Act.

4.2. IMPACTS OF DEREGULATION

An important issue in a deregulated electricity-generating industry concerns the trading framework for the electricity-producing utilities. The US Californian model is described by way of example, see Shiffer (1999). In this model, the generation of electricity is carried out by independent companies (generators) with customers free to choose their supplier of choice. The price that the generators receive can either be negotiated directly with the customer or is determined by a state-owned trading entity called the Power Exchange, into which all generators bid a price for their services. Other countries have similar models; indeed the US approach was borrowed from the UK model.

There are profound financial impacts on nuclear generators arising from deregulation and this *modus operandum* is summarised in Table 4.1 (together with other cultural and

Table 4.1. Impacts from deregulation

Financial	Competitive electricity price All costs must be covered from revenues Variable price for electricity
Cultural	Equal weighting of cost competitiveness, operational excellence and safety Establishment of criteria for this balance Management of employees' concerns over safety implications of new culture
Personnel	Staff concerns about resource reduction, possible relocation, future career path implications, doubts about corporate commitment – new management needs to address Greater employee empowerment and responsibility More modern, efficient processes

Shiffer (1999).

personnel issues). The electricity price received is clearly determined from a competitive bidding process; this price may be reducing. The revenues for the plant must cover the range of costs incurred from operation, including operation and maintenance, capital, fuel and taxes, which are broadly fixed, at least on the timescale of price fluctuations. There is considerable uncertainty on the price obtained, which may vary from hour to hour.

In the UK electricity-generation sector, competition since deregulation has forced down wholesale electricity prices. The UK currently has sufficient generating capacity, but the reduction in nuclear generation and closures of some coal stations may result in smaller reserve margins. New electricity trading arrangements (NETA) have been introduced to encourage flexibility and still further competition. This will alter the market structure and remove payment for capacity; this would tend to penalise baseload generators such as nuclear plants.

Pre-deregulation, the primary objective was high-quality technical plant and operational standards. Achieving low costs was not the highest priority. Post-deregulation, low operational costs must be achieved and these can only be attained through high operational efficiency. The challenge for designers and managers is to ensure that operational efficiency and low cost operation also equate with high reliability and standards of safety. IAEA Technical Report No. 369 provides a description of good practices.

4.3. PLANT MANAGEMENT

The experience from nuclear plant operation has shown that effective plant management and organisational structures are essential to support all operations of plant activity including normal operation, maintenance, refuelling, etc. These are also necessary to achieve the economic performance required by the licensee and to meet the environmental and safety standards required by the regulator.

A good description of practices presently adopted by international bodies and the results achieved is given in IAEA Technical Report No. 369. This book is written in the form of a manual describing a number of ways by which interested parties can transfer to their own situation, the experiences of experts from a number of IAEA Member States. Many of these management practices (developed in the remainder of this section) apply across other large industries.

It requires significant managerial skill to achieve the necessary cost reductions and resource allocations against budget limitations without impacting on the operational and safety performance of the plant. To facilitate these, senior managers must define carefully and communicate to their workforce, the objectives, strategies and criteria in order that correct decisions can be made at all levels to achieve balance between the trade-offs of operational excellence and low cost.

To further the implementation of high standards, an approved quality assurance (QA) programme is usually mandatory, which should be periodically reviewed and updated. International standards such as ISO-9000 are required for many plants (or other comparable quality management systems).

To achieve good operational efficiency, it is important to have a positive and well-motivated work-force and this can only achieved via good employee and management relations. Management must be seen to implement its stated programmes in order to command the necessary respect and support from employees. It must also maintain good relationships with contractors.

In cutting costs to improve efficiency, it may be necessary to reduce staffing levels and this situation may impact adversely on staff morale. Technical Support Organisations can also be impacted if the operating companies place less work outside. This situation must be managed to ensure that the resource reduction is achieved via voluntary staff release as far as possible.

Clearly loss of valuable technical expertise may not result in increased efficiency if there is an inadequate quota of technically trained staff remaining. In many instances, it is the more able staff who leave first during uncertainty and times of change. At a later date, there may be increased costs in recruiting and training new staff in the highly specialised and technical nuclear industry.

Efficiency may be improved via new and more innovative management approaches. New management may consider greater empowerment to lower levels of the organisation. Delays in processes (e.g. signing of routine forms) due to the absence of senior management and the burden on senior management in performing such activities are not efficient. A more imaginative management may introduce simpler work processes.

It is important that the consequences of necessary management decisions to improve efficiency are portrayed in a balanced picture. In times of change, employees may dwell on more negative aspects but for example, reduction in work-force and greater empowerment will mean that the remaining employees have greater responsibilities, more rewarding jobs and more opportunities for promotion. Such positive aspects should be recognised in good management teams.

4.4. OPERATIONAL FLEXIBILITY

The advent of deregulation in many countries brings with it some form of commercialisation and competition in the way that its nuclear plants are now being operated. Part of this culture is to recognise that change becomes a normal way of life. If a nuclear plant utility is to succeed, then the management must recognise and be capable of managing change. Change also brings with it an increased risk, as performance targets are set to stimulate production and new and innovative courses of action are encouraged in the management team to achieve these targets.

A number of important characteristics applying particularly to a change in management system have been defined in IAEA-TECDOC-1123. These include the following, some of which were mentioned earlier.

Organisational changes must be communicated by managers in a way that all levels of staff can understand and accept. Limits of authority must be clearly set. Critical performance variables must be monitored and systems should be in place to do this. Advantage should be taken of latest information and technology (IT) systems to facilitate good communication and feedback to management. There must also be sufficient internal controls and audits to confirm that procedures are being performed satisfactorily.

4.5. OPERATING MARGINS

The plant operating envelope is agreed between the licensee and the regulator as part of the plant safety case. It is usually defined (Pershagen, 1989) by a set of rules and guidelines to ensure safe operation of the plant but also containing some degree of flexibility to enable the plant to operate in an optimal way. The degree of optimisation or the operating margins that can be achieved must be compliant with these rules and guidelines.

They include technical specifications, Table 4.2, which define bounding values for key safety-related parameters. If exceeded, the plant would need to shut down and the regulator would require a full investigation before operation could restart. There are requirements on the functioning of safety systems and components in order that the conditions of plant operation are met. If not all these requirements are met a reduced mode

Table 4.2. Technical specifications

Bounding values for the safety parameters and reporting arrangements to safety authorities if limits are exceeded
Allowable conditions for plant operation, including systems availability – how operations must be restricted if such systems functions are not in place
Specification and schedule for testing and inspection of components and systems – restrictions, if testing is not carried out or functionality is impaired
Rules for both normal and abnormal operation – reporting procedures for operational events and design modifications

Pershagen (1989).

of plant operation may be imposed. Conversely, a more optimised mode of operation may require more stringent performance of the systems and components, possibly a need for plant modifications. Similarly, the degree of optimisation that can be achieved may depend on the outcomes of inspection and testing programmes. Finally, any change in operating conditions must meet the rules for both normal and fault conditions.

The operating rules cover all plant states from start-up to shut-down and in all modes of plant operation. These are documented in detail and may be updated in the light of new experience on changes in plant, e.g. modifications.

4.6. PERFORMANCE OPTIMISATION

The main areas that contribute to nuclear plant availability and reliability have been investigated in an IAEA study of six representative plants (IAEA-TECDOC-1098, 1999; Table 4.3). This involved case studies of plants from Western and Eastern Europe and the US. The plants chosen for the study had exhibited high-energy availability factors and also improvements over recent years in safety, availability and reliability. In addition, the plants

Table 4.3. Representative power plants

Name	Type	Capacity (MWe)	Owner/country	EAF
Blayais	4-unit PWR	3640	EDF/France	78% (1990) 84% (1995)
Trillo	1-unit PWR	1066	Utility Group/Spain	75% (1990) 86% (1995)
Limerick	2-unit BWR	2220	PECO/US	84% (1990) – unit 1 90% (1995) – unit 2
Dukovany	4-unit VVER-440/213	1760	Czech Power Company	82% (1992–1995)
Paks	4-unit VVER-440/213	1840	Hungarian Electric Energy Board	85% (1995)
Wolsong unit-1	CANDU-PHWR	600	KEPC	87% (1990–1995)

IAEA-TECDOC-1098 (1999).

covered different types of water reactors including PWR, BWR, VVER and PHWR designs.

The IAEA study concluded there were a number of management practices that contributed to good plant performance, e.g. organisational structure, strategic planning and objectives, management involvement, internal communication, quality management, relationships with contractors and financial management.

Management philosophy should embody a wide range of core values, which should be conveyed to all its employees. These include diverse aspects including environmental respect, economic competitiveness, the engendering of team spirit and also the characteristics of trust and integrity. It should also include technical aspects such as adherence to ALARA principles and ageing.

Factors that had a direct influence on plant performance included personnel characteristics, the training and development of personnel and the behaviour and attitude of personnel. This conclusion was common for all staff in the workforce, including permanent employees and contractors.

Another conclusion concerned working practices. These included the monitoring of the plant states, the quality of operating procedures, maintenance policy, technical support and interaction and communication between different work groups.

There is general agreement across nuclear plant operating countries, on the necessary working practices for plant performance improvements. These relate to plant status control monitoring, the quality of operating procedures, maintenance policy, technical support and interaction between various informed working groups. The implementation of these practices inevitably varies from plant to plant, depending on local strategies. It is also widely recognised that there are benefits in utilising all levels of local, national and international experience to continue to improve performance.

4.7. MAINTENANCE PRACTICE

Maintenance activities fall broadly into four headings. These include the policy implemented by the plant manager including the balance of maintenance activities and the clearing of backlog activities, the planning and scheduling, the procedures, and the conduct of the maintenance (IAEA-TECDOC-1098, 1999; Table 4.4).

Good availability and reliability are the key objectives. Meeting these objectives requires adequate resources to be available to predict the need for the necessary maintenance, to prevent unnecessary activities, and to ensure maintenance is carried out correctly. The majority of work has to be performed during outages and plant availability depends on it being carried out efficiently and successfully. Unplanned outages should clearly be avoided. In nuclear plants today, computer scheduling systems are used to co-ordinate activities and ensure that adequate materials (spares) are available.

Table 4.4. Effective maintenance factors

Clear maintenance back-logs
Apply maintenance performance indicators
Readiness for eventual unplanned outages
Employ reliability and condition-based decision analysis
Advanced planning of routine maintenance and outages
Use plant-approved procedures in conducting maintenance
Post-maintenance testing to verify satisfactory completion and restart readiness
Use PSA for planning on-line maintenance

IAEA-TECDOC-1098 (1999).

Factors that enhance the effectiveness of maintenance management identified in IAEA-TECDOC-1098 (1999) include, ensuring that maintenance backlog actions do not accumulate and that indicators of various different maintenance activities are recorded. Examples of such indicators include the number of requests, the distinction between preventive and repair maintenance activities and the number of repetitions of work on the same plant components or systems.

On-line monitoring of equipment together with reliability and condition based decision analysis help to reduce preventative maintenance. Preparedness for routine maintenance and outages may be facilitated by advanced planning using mock-ups and other practical demonstrations. These should allow for unplanned outages and should include lists of the different maintenance activities that can be carried out. Regarding technical matters of plant operation, there must be careful monitoring of foreign material in the plant in order to protect equipment. Maintenance should be performed in line with approved procedures, including PSA to facilitate on-line maintenance. Finally, post-maintenance testing must be carried out to verify satisfactory completion of work and to confirm the readiness for the plant restart.

4.8. LOAD FACTORS

Various performance indicators have been defined for measuring the success of a plant in terms of its availability to produce energy safely and economically. The maintenance of good availability depends primarily on the following (IAEA-TECDOC-1098, 1999): control of outage activities, reduction of unplanned outage, reduction of plant transients, improvement of thermal efficiency, good housekeeping of the facilities, minimising plant ageing and optimising staff utilisation.

Different agencies have put forward different performance indicators (IAEA-TECDOC-1098, 1999; WANO, 2002) but they have much in common. Important indicators include, e.g. the energy availability factor (EAF) and the unit capacity factor (UCF) or cumulative EAF. These are included among the IAEA and its power reactor information system (PRIS) measures and WANO measures. The EAF is defined to be the ratio of

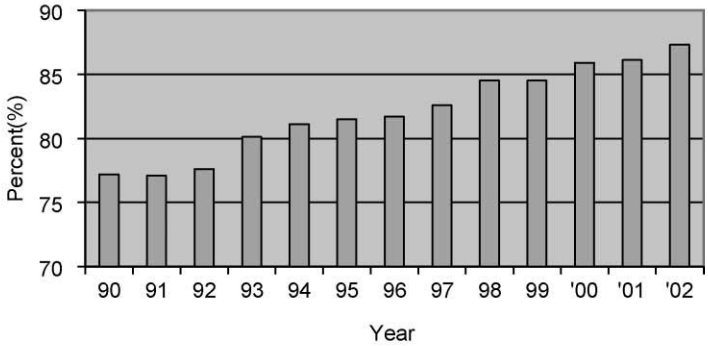

Figure 4.1. Unit capability factor (WANO). Source: WANO (2002).

the actual energy generation (net) in a given period, as a percentage of the maximum energy that could have been produced by continuous operation. Figure 4.1 shows how the worldwide UCF (WANO) has steadily improved over the last decade. Precise definitions of the WANO measures are given in Table 3.1.

Unavailability factors are also considered by both agencies and others. The unavailability factor is usually broken down into planned (PUF) and unplanned energy unavailability factors (UUF) (IAEA-TECDOC-1098, 1999). In IAEA-TECDOC-1098 (1999), energy losses are considered to be planned if scheduled 4 weeks in advance. Planned energy losses include planned outages for refuelling, maintenance, testing, etc. under management control. Unplanned outages include not only unplanned outages requiring similar activities, but also for losses beyond the control of management. Figure 4.2 shows how the unplanned capability loss factor of WANO has steadily reduced over the last decade. Figure 4.3 shows a similar reduction in unplanned automatic scrams.

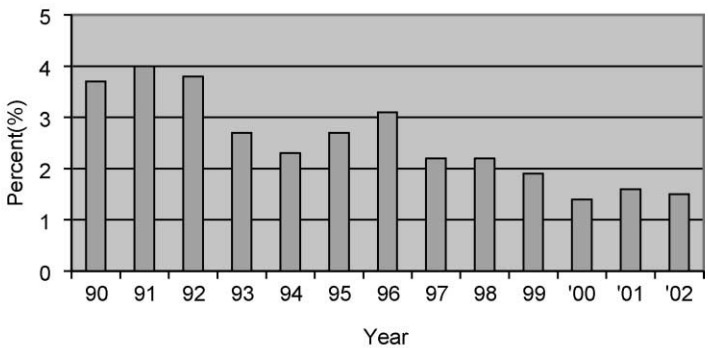

Figure 4.2. Unplanned capability loss factor (WANO). Source: WANO (2002).

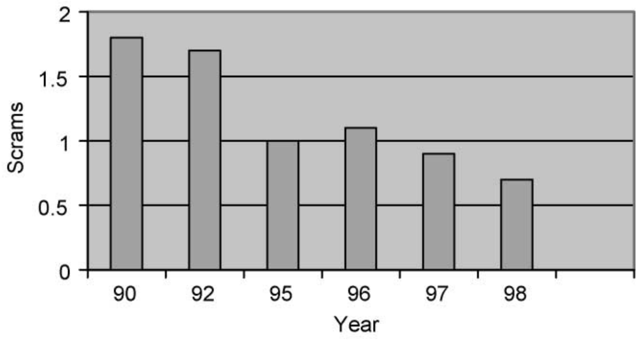

Figure 4.3. Unplanned automatic scrams per 7000 h critical (WANO). Source: IAEA-TECDOC-1175 (2000).

These improvements are attributed to improvement in plant maintenance management and through taking advantage of the benefit of experience.

Across the range of reactor types, the PWR, BWR and AGR units have kept a broadly constant level of performance in recent years. For VVER & RBMK units, there was a decrease in energy availability in the early 1990s, but there has been recovery more recently. The initial decrease was due to the implementation of back-fitting programmes and increase in other maintenance activities during this period.

4.9. RECENT PLANT IMPROVEMENTS

This section provides some examples of recent programmes that are being implemented to improve performance and reliability across a span of the reactors that are currently operating, see IAEA-TECDOC-1175 (2000). In some cases, the improvements are also being incorporated for safety reasons; the resulting better performance is an additional bonus.

In Central Europe, various safety upgrades and reliability improvements have been made on the earlier VVER-440/230 reactors that are still operating. For example, in the Bohunice Units 1 and 2 in Slovakia, the emergency core cooling systems and electrical systems have been reconstructed to achieve better separation redundancy and independence. On each plant, the instrumentation and control (I & C) system has also been reconstructed and an emergency feed water system (EFWS) has been added. Other significant improvements include annealing of the reactor pressure vessels and better seismic qualification.

The improvements for these earlier VVER-440/230s are to ensure safe and economic operation for only a relatively short period of operation; most will be decommissioned within the next few years. Improvements are being carried out on the newer VVER-440/213 reactors to ensure operation for one or possibly two decades into the future.

There have been major modernisation and upgrading programmes on the Dukovany plants in the Czech Republic. Achievement of improved economics by increasing the power rating of each unit by as much as 20 MWe may be possible via an improved evaluation of the operating margins. Modernisation activities include better fire protection, improved I & C systems, modification of the EFWS and better hydrogen control under accident conditions. The Paks plants in Hungary are undergoing similar enhancements.

In Japan, there has been an active programme to establish the necessary inspection and maintenance activities to be done as countermeasures against ageing. There are ambitious targets to extend the life of some of the older plants out to 60 years.

The most modern plants are already incorporating technical features in their design for better practice, improved operation and better maintenance. The latest advanced boiling water reactor (ABWR) plants (Kashiwazaki-Kariwa Units 6 & 7) that entered operation in Japan in the last decade have the largest capacity, yet shortest outage times. This is seen to be due to national component testing programmes to verify their performance for Japanese ABWR operation, even if previous international experience exists elsewhere. Further, during the initial outages, there were rigorous overhauls and inspections of new design features (reactor pumps, advanced control drive mechanisms, high-efficiency steam turbines). There have also been full-scale training programmes for reactor maintenance staff.

4.10. COMPONENT MANAGEMENT AND TESTING

There have been new equipment and techniques developed for component inspection, maintenance, repair and replacement. These have been developed and tested in the laboratory and in full-scale experiments before being applied to plants (IAEA-TECDOC-1175, 2000).

In Japan, techniques are being developed for the chemical decontamination of LWR reactor systems in preparation for the replacement of core internals. For the latter operations, full-scale mock-ups have been used. Techniques for the replacement of a PWR core barrel and bottom mounted instrumentation systems are being examined. Methods for the replacement of BWR core housing, core shroud, control rod housing and jet pump riser braces are also under consideration.

Holographic methods of inspection for the recognition and sizing of cracks are being developed. In Japan, intergranular stress corrosion cracking has affected reactor internals and core shroud and ways are being evaluated for mitigating this phenomenon. One solution is to replace components fabricated with SS type 304 with corresponding components made of SS type 316.

Manipulators are being developed for the purposes of in-pipe inspection, grinding and for repairing cracks in welds, e.g. between the vessel nozzles and pipes.

4.11. REACTOR SURVEILLANCE AND DIAGNOSTICS

Instrumentation methods and models for core monitoring are being developed and validated to provide detailed and more reliable information on local core power and other parameters affecting the fuel duty, e.g. the BWR core decay ratio (Vattenfall AB/ABB Atom/Swedish Nuclear Power Inspectorate, 1999). Core monitoring systems can also be used to support reactor operation under normal conditions and transients. They can also provide data on initial core parameters that are required for transient and accident analysis. These methods are being developed for all LWRs.

The Stockholm conference (Vattenfall AB/ABB Atom/Swedish Nuclear Power Inspectorate, 1999) addressed the main areas of current activity including the requirements for core monitoring systems and sensors, improvements in these systems, signal processing and evaluation, and design and operating experience. It also covered improved core models in core monitoring. Major trends observed include the availability of better physics models for on-line calculations for both PWR and BWR systems. There is a continuing development of improved physics techniques that will result in improved models being implemented in core monitoring systems. Information from on-line measurements is being combined with on-line calculations, enabling further model validation. There may be advantages of back-fitting some older PWRs with fixed in-core detectors.

Reactor surveillance and diagnostics issues have been the subject of various SMORN conferences, see 7th Symposium on Reactor Surveillance and Diagnostics (SMORN-VII) (1995). These conferences have covered different themes including techniques of surveillance and diagnostics in reactor primary circuits, including the sensor itself and the techniques for processing information. Other areas included have been the validation and surveillance of sensors, acoustic leak detection, thermal–hydraulic measurements and the detection of boiling. Methods for the detection of loose parts, vibration of structures, experience of reactor operation and the performance of their surveillance systems have also been addressed. Latest techniques included advanced signal processing and the use of neural networks. Monitoring of BWR stability, estimation of particular physical parameters, neutronics studies and analytical techniques are also on-going subjects of study.

4.12. OUTAGE MINIMISATION

Reduction in outage time has been achieved in a number of plants that have been operating for some years through both technical and administrative improvements. Another factor has been the introduction of more computerised systems to aid in the planning and managing of outages. Technologies for improving light water operation and maintenance,

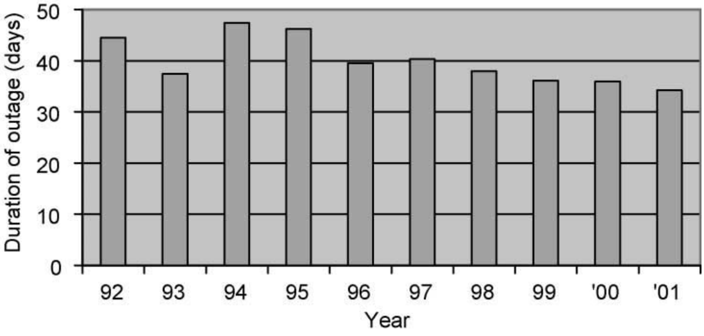

Figure 4.4. Global average time for planned outages. Source: IAEA Technology Annual Report (2002).

including outages, have been published in IAEA-TECDOC-1175 (2000). Figure 4.4 shows the global reduction in average planned outage time that has been achieved over the last decade.

The fuelling scheme clearly has an important influence on outage planning. For example, there is a general tendency towards longer fuel cycles. There are obvious requirements on the load design, the most important of which is safe operation, e.g. limits on shutdown margin, and ensuring negative moderator temperature reactivity coefficient. Once safety constraints have been satisfied, outages may need to be synchronised in order that there is a period of time between outages of different units at the same plant. Clearly outages should be avoided at times of peak demand, e.g. during winter periods, etc. Finally, outages should be planned in order to optimise fuel economy and cost.

Outages may be of different duration depending on the work planned. In-service rules may require complete in-service of the reactor vessel including complete unloading of the fuel, the removal of reactor internals, inspection of the reactor vessel at regular intervals, e.g. every 4 years. Shorter outages would be used to carry out a less ambitious programme. By way of example, in the Paks plant in Hungary, the new outage strategy includes outages of short (25–30 days), normal (30–35) and long (55–60 days) duration.

Another example (IAEA-TECDOC-1175, 2000) where a different technical approach has resulted in improvements of outage time concerns the Indian Point 2 reactor in the US. Here two safety systems modifications have been implemented, replacement of conventional hydrogen ignition systems with passive auto-catalytic hydrogen recombiners and the replacement of the conventional containment spray additive tank with baskets of tri-sodium phosphate. The original systems consisted of several hundred components which required significant maintenance and testing, the new passive systems require much less.

For the future, ways to reduce outage are being considered at the design stage, e.g. in the advanced European pressurised reactor (EPR) design. These include features such as improved accessibility of the reactor building during operation, and better logistics support including the need for special tools and availability of spare parts. Some of these approaches are also being considered for current plant.

4.13. ADVANCES IN DESIGN AND TECHNOLOGY

In current generation and new plants, digital instrumentation, control technologies and also self-diagnostic systems are under development. There are also new control room and man–machine interface improvements that include human factors engineering considerations. Examples of reactors with improved control room design include the latest PWR designs under consideration in Japan and also the control room design in the Korean next generation reactor.

Many of the improved practices in regard to design and technology of new plants and in the back-fitting of older plants comply with the traditional design basis objectives to include increased redundancy and diversity (IAEA-TECDOC-1175, 2000). Thus more emphasis is placed on reducing vulnerability to single component failure and in ensuring that the design accommodates sufficient scope for maintenance during plant operation. Increased diversity reduces the frequency of common mode failure. The VVER-440/230 improvements referred to in Section 4.9 are a good example of back-fitting improvements resulting in improved redundancy and diversity. Other practices include the careful planning of the plant geography to ensure access for inspection, maintenance, replacement and repair; these are being studied in Japan as are the increased use on-line testing and maintenance practices, referred to in Sections 4.9 and 4.10.

4.14. NUCLEAR FUEL CYCLE

Improvements in nuclear fuel technology and the fuel cycle are leading to better performance and economics of power plant operation. These are considered in the next chapter.

REFERENCES

IAEA Technical Report No. 369, ISBN 92 0 193184 7. Management for excellence in nuclear power plant performance.

IAEA-TECDOC-1123, Strategies for competitive nuclear power plants.

IAEA-TECDOC-1098 (1999) Evaluating and improving nuclear power plant operating performance, July 1999.

IAEA-TECDOC-1175 (2000) Technologies for improving current and future light water reactor operation and maintenance: development on the basis of experience, September 2000.
IAEA Technology Annual Report (2002) IAEA web site: http://www.iaea.org.
Pershagen, B. (1989) *Light Water Reactor Safety*, Pergamon Press, New York.
Shiffer, J.D. (1999) Issues for nuclear plants in a de-regulated electricity supply industry. *Nucl. Energy*, **38** (4), 259–264.
7th Symposium on Reactor Surveillance and Diagnostics (SMORN-VII), Avignon, 19–23 June 1995, http://www.nea.fr/html/science/docs/.
Vattenfall AB/ABB Atom/Swedish Nuclear Power Inspectorate (1999) Reactor surveillance and diagnostics workshop on core monitoring for commercial reactors: improvements in systems and methods, Stockholm, 4–5 October 1999, http://www.nea.fr/html/science/rsd/.
WANO (2002) Performance indicators, http://www.wano.org.uk.

Chapter 5
Nuclear Fuel Cycle

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Chapter 5

Nuclear Fuel Cycle

5.1. INTRODUCTION/OBJECTIVES

Within present day generation plants, there is a continuous drive to improve all aspects of performance and safety in nuclear fuel cycle technology and practices. The approach is to optimise fuel cycles as a whole, taking into account all components from mining to disposal. This will include the various options for fuel supply, fabrication, generation, fuel storage, reprocessing, recycling, waste management, disposal and decommissioning. It will be necessary to simplify the fuel cycle to reduce costs, while still minimising the environmental impact, maintaining the safety of and retaining public confidence in fuel cycle facilities. Particular goals are to improve fuel performance for longer life in the reactor and the development of advanced fuels, including mixed oxide (MOX) fuel. This chapter reviews these issues and practices in turn.

There are many drivers for advanced fuels development. For current and evolutionary plant optimised uranium-based fuels are being considered to enable higher power, longer life and longer fuel cycles. The utilisation of MOX fuels in thermal reactors is one method of burning unwanted plutonium from weapons programmes. Other fuels such as thorium offer advantages of reduced actinide inventory in waste. Future reactor systems offer a means of managing actinides and reducing the radiotoxicity of waste.

5.2. FUEL CYCLE OPTIMISATION

Fuel cycle strategies should satisfy a range of criteria for optimising performance (Ion and Bonser, 1997). Clearly they should aim to utilise the available fissile material in full. This might be achieved by recycling of uranium and plutonium from already irradiated fuel or through other sources. The total fuel cycle costs must also be optimised in order to maximise the utilities' performance from an economic perspective. There is increasing tightening of regulations from national governments and international bodies such as the EC on environmental releases. The impacts of fuel cycle operations on the environment should be minimised by optimised waste and spent fuel management planning. Since fuel cycle activities inevitably involve the handling of fissile material through reprocessing or other means; the political and proliferation issues must also be adequately managed.

The notion of a holistic fuel cycle has been put forward by BNFL; the main elements are summarised in Table 5.1. The holistic approach recognises that different systems and fuel cycle policies can exist in various countries but it is sufficiently flexible to accommodate

Table 5.1. Holistic fuel cycle

Integration of	Requirements at each stage
Fuel fabrication	Maximising safety
Electricity generation	Minimising waste
Reprocessing	Minimising cost
Used fuel products	Security
Waste management	Safeguards
Disposal	
Decommissioning	

Ion and Bonser (1997).

such differences. The approach has been adopted for current fuel cycles, e.g. for the AGR fuel cycle in the UK, for MOX fuel cycles in LWRs (light water reactors) and for LWR fuel in CANDUs. The approach is also being adopted for advanced fuel designs.

5.3. FUEL RESOURCES AND SUPPLY

There are sufficient high-grade uranium reserves to service the present fleet of reactors for at least 60 years based on the present fuel strategies and a demand of 65,000 tonnes per year (Energy Visions 2030 for Finland, 2003). Since the price of uranium has been dropping from the early 1980s, the emphasis of the mining industry has been to concentrate on the high-grade resources. There are considerable additional reserves anticipated from undiscovered conventional deposits and even more from less conventional sources such as seawater (Table 5.2).

Low-cost uranium resources are distributed worldwide as shown in Figure 5.1. In terms of production, the largest producer is Canada, generating about one-third of the total world supply. The next largest is Australia, about one-sixth, and other significant contributions come from Nigeria, Namibia and Russia.

Current production is around half of demand with the remainder coming from uranium stockpiles for the civil nuclear programmes in the US and Russia. There are considerably

Table 5.2. Uranium resources

Reserves	Amount (M tonnes)	Total fuel provision time (years) based on current fleets' usage and fuel cycle strategies
Present known high-grade reserves	4	At least 60 years
Undiscovered conventional deposits	11	~250
Unconventional deposits, e.g. sea water	22	

Data supplied by Energy Visions 2030 for Finland (2003).

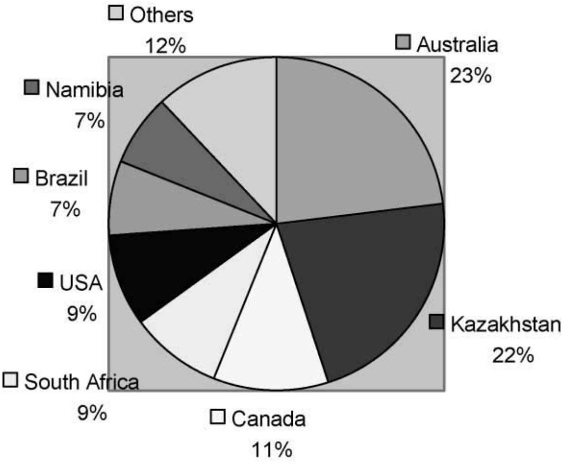

Figure 5.1. Country distribution of high-grade uranium reserves. Source: Energy Visions 2030 for Finland (2003).

more supplies available from highly enriched uranium and plutonium from dismantled weapons.

Regarding the efficient utilisation of fuel, a fast breeder reactor fuel cycle would use the uranium 50 times more efficiently, compared with other fuel cycles.

A thorium cycle would result in further fuel resource; thorium fuel is four times more abundant than uranium.

5.4. ENRICHMENT

Current enrichment technologies are based on either gaseous diffusion or centrifuge methods. Other methods based on curved nozzle separation and laser enrichment have also been explored (Leclercq, 1986). The capital costs of enrichment plants are relatively high; around 6% of the total generation cost (Bertel and Wilmer, 2003).

Gaseous diffusion is a widely used method of enrichment in many countries. In this process, uranium hexafluoride is enriched by diffusion through porous barriers. The process is repeated through a large number of stages until the required enrichment is obtained. Capital costs are therefore high. The engineering issues relate to constructing corrosion resistant and efficient barriers to prevent blocking of the pores. For the process to work efficiently, feed streams need to be compressed and then the heat of compression removed to maintain the gases at the correct temperature and pressure. This is an energy-intensive process and therefore gaseous diffusion plants have high operating costs because of their large electricity requirements.

In centrifuge plants the uranium hexafluoride gas is spun in a vertical centrifuge and the U-235 concentrated near the axis. The high rotational speeds place limits on centrifuge capacity and therefore many (thousands) of the identical centrifuges are required. Centrifuge plants are, therefore, also capital intensive. However, operating costs are less than for gaseous diffusion. The gas must be successively centrifuged in stages but the number of stages is approximately 10 times less. Also the operating energy requirements are 10 times less than for gaseous diffusion plants.

At the present time, there is a surplus of enrichment capacity in operation in the world, despite the fact that few companies operate enrichment plants and the number of plants is very small (Bertel and Wilmer, 2003). However some plants have been in operation for 25 years and will need to be replaced. The next generation of enrichment plants is likely to be based on centrifuge technology.

5.5. FABRICATION

The fuel design and fabrication process is clearly very dependent on the type of nuclear power plant in question. Only some gas cooled plants (e.g. Magnox) and some small reactors still use uranium metal fuel; all other types use uranium (or possibly mixed) oxide fuel.

Fuel design and fabrication costs are relatively small at about 3% of the nuclear electricity cost (Bertel and Wilmer, 2003). However, fuel design and fabrication have an important influence on the successful operation of the plant. There is strong competition among fuel vendors to achieve better utilisation and higher plant availability. Fuel designs to increase the discharge burn-up are an objective of current fuel vendors. Competition between vendors is also heightened since at the present time there is LWR manufacturing capacity around 50% in excess of annual requirements.

5.6. FUEL DESIGN

A good review of current and future fuel cycle options for LWRs and HWRs (heavy water reactors) is given in IAEA-TECDOC-1122 (1998) (Table 5.3).

5.6.1 Light Water Reactors

5.6.1.1 Present. LWRs are the most widely operating type of reactor in the world and LWR fuel optimisation is of international interest. There is intense competition between fuel vendors and there are many different designs offering different performance advantages. However, there has been extensive experience amassed on fuel performance, and fuel designs based on a conventional uranium cycle are well optimised. Thus the

Table 5.3. Advanced fuels options

Type of fuel	Plant
MOX	LWR, HWR
Thoria fuel	LWR, HWR
Inert matrix/uranium free fuel	LWR
Slightly enriched uranium	HWR
Recycled uranium	HWR, LWR (with enrichment)
Fuel for direct recycle	HWR
Ceramic fuel	HTR
Fuel cycles for plutonium and minor actinide destruction	FR

Data from IAEA-TECDOC-1122 (1998).

differences in such designs are relatively small. There has been even some measure of collaboration between fuel vendors arising from the need to share costs associated with expensive research programmes. The common drivers in fuel design are to achieve greater reliability, to reduce fuel failures, to move towards higher burn-up and to reduce fuel cycle costs. These fuel performance issues are considered in the Section 5.6.1.2.

In addition to the conventional uranium fuel cycle for LWRs, MOX fuel has also been used and is well established. In the MOX fuel cycle, plutonium oxide is mixed with uranium dioxide for use as fuel in LWRs. MOX fuel is used in France, Germany and Japan. It was first used in Europe and the US in the mid-1960s and since then hundreds of tonnes of MOX fuel have been burnt in commercial LWRs. The success of burning plutonium in MOX fuel demonstrates that plutonium is an asset that can be used for civil nuclear power generation. Further this has been realised by the development and safe operation of large-scale plutonium recycling facilities in France and most recently in the UK, now that the BNFL Sellafield MOX plant has become operational. The IAEA have put in place controls to ensure adequate safeguarding of materials.

5.6.1.2 Future

MOX. MOX fuels represent the most significant developments in LWR fuel technology, particularly in Europe (IAEA-TECDOC-1122, 1998). MOX fuel up to 30% loading can be used in LWRs within current operating and safety margins; higher percentage loadings would require control rod changes to maintain current margins. MOX fuel costs are higher than UO_2 fuel costs but this largely reflects reduced production at the present time. The potential of advanced MOX fuel is being studied in France and Japan.

CEA are investigating advanced plutonium fuel assemblies to overcome the problems of multiple plutonium recycling in PWR MOX assemblies. As MOX assemblies are irradiated, the isotonic quality of the plutonium is reduced (Groullier, 2001). CEA are working on high moderation plutonium fuels (Youniou *et al*., 1998). In conventional MOX assemblies, the moderator/fuel volume of MOX is the same as in UO_2 assemblies and new designs are being investigated to increase this ratio which gives a more complete thermal flux and reduces the initial plutonium content. Conversely, the Japanese are looking to lower moderation fuels to achieve plutonium breeding, see for example Tochihara *et al*. (1998).

Thoria Fuel. The development of thoria fuel has been overshadowed by the emphasis and investment in uranium-based fuel. Nevertheless, thorium is about three to four times more abundant than uranium and represents a good long-term nuclear fuel supply. The cycle produces fissile U-233, thereby enabling breeding potential in a thermal reactor, good in-core behaviour and lower excess reactivity requirements. A disadvantage is that thorium ore does not contain a fissile isotope and so U-235 or Pu must initially be used in conjunction. Thorium fuel is attractive for various reasons. There is very little production of plutonium or transuranics, which reduces the radiotoxicity burden and, therefore, there is a benefit from the point of view of proliferation (Hesketh, 2003). Thoria fuel has been successfully demonstrated in power reactors.

Uranium Free Fuels. The incineration of plutonium from weapons programmes and from reprocessed LWR fuel is under consideration in many countries. Another pressing issue to the nuclear countries is how to burn actinides as part of a waste management strategy. Research programmes are underway in Switzerland, Japan, France and Canada. The idea is to burn the plutonium (or actinides) in a non-fissile inert carrier matrix. Various fuel matrices are being examined, e.g. zirconium oxide in Switzerland, fluorite and spinel in Japan, ceramic (spinel, magnesia) or metallic matrices in France and silicon carbide (SiC) in Canada. Other materials may also be required, burnable poisons (e.g. erbium) for control of reactivity and addition of thorium or uranium to enhance negative temperature coefficient. To date, fuels have largely been irradiated with accelerators; initial results are good for SiC and zirconia. Some in-reactor irradiations have taken place. The main issues relate to materials performance that are not yet resolved and inert matrix fuels are unlikely to enter LWR fuel cycles in the near future.

5.6.2 Heavy Water Reactors

5.6.2.1 Present. Much of the discussion above for LWRs in terms of drivers for fuel design to achieve greater reliability, to reduce fuel failures, to move towards higher burn-up and to reduce fuel cycle costs, applies also to HWRs.

5.6.2.2 Future

Slightly Enriched Uranium. Slightly enriched uranium (SEU) fuel of 0.9% U-235 enrichment is a promising fuel for HWR CANDU reactors (IAEA-TECDOC-1122, 1998).

With such an enrichment, burn-ups of about 13,800 MWd tU^{-1} can be achieved, an improvement by about 34% on uranium resource allocation compared to that achievable with natural uranium. An additional advantage is to reduce the volume of spent fuel produced and to reduce fuelling costs. It can also be used for power upgrade by flattening the radial power profile. Several fuel management schemes have been investigated by Atomic Energy of Canada Ltd (AECL) using the 43 element CANFLEX fuel design in a CANDU 6 reactor. These show greater fuel performance margin compared with the 37 element design because the CANFLEX design results in peak fuel linear ratings about 20% lower.

Recycled Uranium. Recycled uranium (RU) from the conventional reprocessing of spent LWR fuel has an enrichment of 0.9% U-235 and, therefore, can be used directly within the SEU fuel cycle. Therefore, RU can be used directly in HWRs or it can be enriched and used in LWRs. There are significantly large quantities of this material at each reprocessing plant, in particular large amounts of RU have been produced from reprocessing operations in Europe and Japan, around 25,000 tonnes. Further RU has come from reprocessing in the former Soviet Union (FSU). There are economic advantages of using 0.9%-enriched uranium in CANDU reactors. It is possible to increase unit output by over 10% and the average discharge burn-up can be increased by more than 30% (Meneley, 1998).

Fuel for Direct Recycle. For the near future, Canada and Korea are investigating the possibility of burning LWR fuel in CANDU reactors. This utilises the DUPIC process, see below, using a dry reprocessing route and offering a direct route for LWR fuel conversion to fuel for CANDUs. This initiative is being pursued by AECL and the Korean Atomic Energy Research Institute (KAERI).

The direct use of spent PWR fuel in CANDU (the DUPIC process) is a dry reconstitution process for converting LWR pellets into HWR pellets which avoids the separation of U and Pu. The concept takes advantage of the high neutron economy of CANDU reactors enabling the burning of spent LWR fuel in HWRs. There are a number of benefits of this concept. Since it is based on a 'once-through' cycle there is a saving of uranium and also a reduction on the amount of spent fuel to be disposed. It is also attractive from the point of view of non-proliferation since plutonium is not separated during the process. The technology is being developed in Canada and also in Korea where both PWR and CANDU systems are operated. Consideration has also been given in the US to reviving the 'AIROX' dry recycle process for conversion of LWR to LWR fuel, a process similar to DUPIC.

MOX. For HWRs, assessments in Canada show there is no barrier to 100% loading. Normal CANDU power densities can be sustained with burn-ups three times those of natural uranium with comparable performance to that of UO_2.

Thoria Fuel. Thorium fuel designs have been considered in Canada within the CANFLEX bundle design, using ThO_2 and SEU elements and also in a novel Indian HWR design. Thoria fuel is well suited to HWRs. Only once through thorium cycles have been considered thus far, recycling requires extensive research and development.

5.6.3 Gas Reactors

5.6.3.1 Present

Magnox and AGR. The objective of UK fuel cycles has been to maximise energy production, while minimising costs and effects on the environment. To this end UO_2 is recycled in AGRs and the AGR and Magnox fuel cycles have been harmonised. About 15,000 t of reprocessed Magnox uranium has been recycled, re-enriched and used in the production of 1500 t of AGR fuel (Ion and Bonser, 1997). Considerable experience has therefore been amassed on manufacturing fuel from recycled spent fuel.

5.6.3.2 Future

HTR. High-temperature reactors have been studied for many decades and are now seen as a possible alternative to evolutionary LWRs. Two designs being considered at the present time are the pebble bed modular reactor (PBMR) and the gas turbine-modular helium reactor (GT-MHR) which utilise a ceramic fuel. This technology was established in earlier prototype plants (Hesketh, 2003).

HTGR fuel particles usually consist of uranium or plutonium oxide, spherical in shape. There is in addition a low-density buffer zone to allow for accumulation of fission gas. This is coated with three coatings, an inner pyrocarbon layer, a silicon carbide layer and an outer pyrocarbon layer. These form a corrosion-resistant barrier to fission product release. The fuel particles are about 1 mm diameter and each layer is about 40 μ thick. These type of fuel particles are known as TRISO fuel.

Experience has been gained from the operating experience of material test reactors and operating power reactors on the limits of fuel behaviour. The current design limit for maximum fuel temperature for normal and abnormal conditions is 1600°C. Tests have shown (Methnani, 2003) that that there is only a small 10^{-4}–10^{-5} probability of particle failure for burn-ups of up to 15.6% fissions per metal atom. At higher temperatures, post-irradiation heating experiments have shown only small fission product release up to 2200°C for heating periods up to 500 h but more significant increases for burn-ups of around 8%. Research is in progress to understand further the mechanisms leading to coating failure and to increase the temperature operating envelope.

There are differences in the way different HTGR designs incorporate the fuel particles within the fuel matrix. In PBMR, the particles are in a graphite matrix in the geometry of 6 cm diameter spherical pebbles that pass continuously through the core. At any given time there are hundreds of thousands of these pebbles within the core region. In the GT-MHR design, the fuel particles are contained in a graphite matrix in cylinders 13 mm in diameter by 51 mm length. About 3000 of these are composed to form a hexagonal graphite fuel element.

The fuel cycle for the pebble bed designs is based on on-line refuelling in which the fuel pebbles pass continuously through the core. Fuel economics can be improved by a multi-pass system in which the pebbles are recycled. This helps to give a flatter power profile and improves fuel utilisation. In contrast, the prismatic design requires a fuelling strategy with periods for outages every few years.

Current fuel design research is focussing on the manufacturing process. The main issues are whether the fuel microspheres are sufficiently reliable; also the economics will be improved if designs can be developed to achieve high burn-ups.

There is a potential to include fertile fuel such as thorium within the TRISO design to improve the fuel management. Since the fuel can be plutonium-based, there is also a potential for the HTGR concept to be used for reducing the stockpile of weapons plutonium. This is because the HTGR has a desirable power spectrum; in principle, thorium could be used as a fertile material as in the earlier Fort Saint Vrain HTR design, plutonium could be burnt within a GT-MHR design concept.

5.6.4 Fast Reactors

The development of fast reactors for electricity generation is largely in abeyance except in Japan and Russia. Work is, however, progressing on the use of fast reactors for consumption of excess plutonium and the destruction of minor actinides (MAs) and long-lived fission products (LLFPs). However, a number of fuel cycle options are being investigated, e.g. in the CAPRA-CADRA programme (Hesketh, 2003).

5.6.4.1 Plutonium Burning Fuel Cycles. Fuel cycles based on a fast reactor plutonium breeding cycle are being investigated for possible adaptation to plutonium consumption. In the CAPRA-CADRA project the European fast reactor (EFR) concept is being considered whereby the plutonium content of a MOX fuel assembly is increased at the expense of U-238; this, therefore, results in net plutonium destruction. Inert matrix assemblies have also been considered based on a plutonium nitride cycle.

5.6.4.2 Minor Actinide Target Fuels. Fuel cycles that produce and burn equal amounts of plutonium and which can destroy MAs and LLFPs are the subject of advanced fuel research. To do this, conventional fast reactors are possible but accelerator driven systems (ADS) may have some advantages. The latter are considered later in the book. From the view point of target fuel assemblies both designs are similar. There are essentially two approaches to fuel design:

homogeneous – where the MA or LLFP is mixed with the fuel, and
heterogeneous – where there is a separate target assembly.

The target materials being considered include oxide, nitride and cermet (ceramic/metal) fuels; of these, nitride fuels are the most promising. There are many options that are being researched, see for example (Hesketh, 2003).

5.7. FUEL PERFORMANCE

The subject of fuel performance is of international importance in the nuclear industry. It has attracted significant attention in all the major countries operating nuclear plant, particularly in France, Germany, Japan, US and the UK.

Good fuel performance is a necessity for utilities and the major advances are driven by the utilities requirements. For example for LWRs, as discussed in the previous section, the use of MOX fuel and the potential to use RU from reprocessing is a particular current interest.

In the UK, within its gas reactor programme, much experience has been gained from many years of successful Magnox and AGR operation. The fuel performance in AGRs has been very good in matching the performance of most other reactor types. Similarly experience on fast reactor fuel performance has been gained in the UK and France; this has helped to support the fuel design for the EFR.

5.7.1 Light Water Reactors

A very important requirement for currently operating plant is reliability in fuel performance. Loss of generation through fuel failures, shutdown, etc. are extremely expensive to an operating utility, of the order of $1 m per day loss of operation. Similarly, inadequate fuel performance that requires down rating of the reactor results in corresponding loss of revenue. Such economic penalties far outweigh gains that might be realised by different fuel designs, if reliability is compromised. For this reason, utilities are generally conservative in changing fuel designs; rod failure rates are now very small, rates typically less than 1 in 10^5 per fuel cycle are currently being achieved. A good review of the important technical and economic factors in regard to fuel-related issues is given in (Hesketh, 2003).

Fuel burn-ups of 60 MWd kgU^{-1} for PWR and 45 MWd kgU^{-1} for BWR are now being achieved (Stanbridge and Howl, 1992). New cladding alloys are under development, an important design requirement being to reduce waterside corrosion of the cladding. Higher burn-up and longer fuel cycles, e.g. cycle lengths increase from 12 to 18–24 months can be achieved via the use of integral absorbers. Longer fuel cycles result in higher fuel costs, due the higher initial enrichment required, but these are more than compensated for by reduced expensive refuelling outages. Thus reduced fuel cycle costs can be achieved by increasing fuel burn-up and longer cycles.

It may also be possible to relax some plant-operating restrictions, if it can be demonstrated that fuel failure probabilities under normal and fault conditions can be reduced by improving fuel designs, cf. the pellet clad interaction (PCI) resistant fuel that is being developed for BWRs.

A good overview of the principal issues is given in (Stanbridge and Howl, 1992). Technical features to improve performance include the following.

There has been a move to increase fuel assembly size for given assembly rating and thus to reduce linear fuel ratings. Assembly designs for PWRs have been increased from 17×17 lattices to 18×18 lattices, for BWRs there has been an increase from 9×9 to 10×10 designs. This results in reduction of fuel temperatures, fission gas release and the propensity for PCI during transients. Fuel rods have been pressurised above ambient pressure to reduce thermal feedback from fission gas release, to slow cladding creep and to also reduce PCI in transients. Other features are the development of more corrosion-resistant cladding via changes in the composition and conditioning of the cladding material and the move to Zircaloy grids to reduce the absorption of neutrons (and this also reduces operator dose through elimination of cobalt present in other grid materials). Debris filters have been introduced in lower assembly nozzles to reduce debris in the coolant and reduce fretting. Integral burnable poisons have been introduced such as gadolinia doping of the fuel pellets and zirconium di-boride coating of fuel pellets resulting in reduction of power peaking.

It is clear that despite the maturity of fuels research, there is a strong push to improve further LWR fuels, to achieve higher burn-up at reduced fuel cycle costs. The drive is to introduce MOX and recycled UO_2 fuel into European and Japanese LWRs and to achieve the goal of very small (zero) failure rates.

Consequently there are significant research programmes devoted to LWR fuel performance. These include in-reactor experiments, i.e. the OECD Halden Reactor Project in Norway and the Nuclear Fuel Industries Research (NFIR) Group run by EPRI in the US. Many countries participate in the Halden project, from Europe, Japan and the US. Data have been obtained from tests for PWR and BWR conditions; there are also some relevant data for AGR fuel. The NFIR Group has instigated activities on basic research on fuel and cladding. Other separate effects tests, power ramp experiments at Studsvik in Sweden, fission gas release experiments at Riso in Denmark have also taken place. Another major programme especially addressing high burn-up fuel is the OECD CABRI project in France.

5.7.1.1 High Burn-Up Fuel Issues. LWR fuel performance at high burn-up has attracted increasing attention in the last decade. Regulators have imposed limits on peak rod burn-up because of concerns on the integrity of the fuel. For example, USNRC limits are about 62 MWd kgU^{-1} on peak fuel burn-up (MacDonald *et al.*, 1998). There are also limits on enrichment (typically limits less than 5%). These are due to limitations on the design and licensing of fuel fabrication plants and other ancillary equipment. There are further limitations on control rod worths and the neutronics design of the core that may limit the use of higher burn-up fuel.

The technical issues regarding the use of LWR fuel at higher burn-up include loss of cladding ductility and fracture toughness due to both chemistry-related and physical-damage-related mechanisms. These include excessive corrosion, hydrogen uptake and zirconium hydride formation, damage due to neutron radiation fluence, oxide spallation and zirconium hydride blister formation. Other cladding changes at higher burn-ups include significant cladding growth.

The main reasons for selecting Zircaloy as the clad material for most LWRs, were because Zircaloy has a small neutron cross-section and relatively good resistance to corrosion (provided that water chemistry is carefully controlled), at least for moderate burn-ups under normal operating conditions. Nevertheless regarding high burn-up performance, examination of Zircaloy clad fuel rods, particularly PWR rods irradiated up to 50–60 MWd kgU^{-1}, has showed that thick oxide layers ($\cong 160\ \mu m$) exhibit spalling. Moreover, Zircaloy blisters formed at some locations, and the remainder of the cladding wall had little ductility or toughness due to the formation of zirconium hydride platelets.

The ductility of Zircaloy is substantially reduced by neutron radiation fluence. The total plastic elongation at burst of Zircaloy tubes under irradiations of 10×10^{21} n cm^{-2} may be as low as 0.5–1% compared with 15–20% for un-irradiated material (MacDonald *et al*., 1998). Zircaloy cladding and other Zircaloy structural materials may also bow which could become a safety issue if the insertion of control rods is affected.

Higher burn-up increases the propensity for increased fuel pellet cladding mechanical reactions (PCMI) due to fuel swelling, cladding creep-down and fuel-cladding diffusion bonding. Such failures have occurred in some LWRs. This has occurred especially in BWRs where there may be significant changes in power associated with control rod movement.

Other phenomena resulting in fuel irradiation at high burn-up are reduced fuel thermal conductivities resulting in increased fuel temperatures. Increased fuel temperatures also result from plutonium and fission production near the surfaces of the fuel pellets and the formation of a porous rim. The fission product production also gives rise to increased fuel rod internal pressures at long irradiation times due to increased time of diffusion.

In order to overcome these problems, fuel vendors have invested significantly in developing improved cladding materials, over the last 20 years. This has resulted in new clads which are more resistant to corrosion, hydrogen uptake and PCMI, than standard Zircaloy. For example, the low tin ZIRLO cladding material only exhibits about one-fourth of the level of corrosion of standard Zircaloy (MacDonald *et al*., 1998). There are a number of other cladding materials which show improved characteristics at higher burn-up showing in addition to improved corrosion resistance, less growth and creep under irradiation. Cladding liners have also been considered for BWRs to provide better protection against PCMI.

The majority of effort has focussed on cladding research rather than the development of new fuel forms. In respect of the latter, there have been relatively minor changes in the pellet diameter to length ratios, density, and grain size to reduce PCI, fuel densification and gas release characteristics. There are some differences in rod geometry between vendors; different rod designs have different plena. More work could be proposed to improve fission production, to result in a more uniform rod internal pressure profile and to minimise PCMI.

5.7.1.2 MOX. Operational experience is being gained on the performance of PuO_2/UO_2 MOX and re-enriched depleted uranium fuel, in comparison with standard UO_2 fuels. For the latter comparison there appears to be little impact on performance, for MOX, fuel performance data are now coming forth from various international programmes (Table 5.4).

The performance of MOX under accident conditions beyond burn-ups greater than 60 MWd kgU^{-1} requires verification (in the same way as does standard UO_2 fuel).

5.7.2 Heavy Water Reactors

Much experience has been gained on HWR fuel performance and various advanced fuel design options are under study (IAEA-TECDOC-984, 1997), described above. This includes the use of SEU (0.9–1.2%) to give economic advantages in terms of higher burn-up (7–11 and 22 MWd kg^{-1} NU, respectively) and less waste handling. These options are being studied at AECL in Canada. Combinations of natural uranium and a SEU core have been studied in different core designs in Argentina.

Table 5.4. MOX fuel characteristics compared with UO_2

Issue	Characteristic	Comparison
Operational: reactivity behaviour with burn-up	Higher local peaking at low burn-up but with margins unaffected	Increased clad corrosion and fission gas release at higher burn-up relative to UO_2
Safety evaluation	Consequences of limiting transients, although more severe with recycled MOX are acceptable	Consequences of some transients/accidents may be less severe with weapons Pu relative to recycled Pu
Fuel performance	At low burn-up, no significant difference in corrosion behaviour; fission gas release higher, but accelerated release only starts at 40 GWD/MTHM	Higher power and higher temperature in MOX at high burn-ups relative to UO_2

Data from Malone *et al*. (1998).

The LWR/CANDU tandem cycle can be achieved by the re-use of spent LWR fuel. This leads to significant additional energy yield (77%) through the use of reprocessed LWR fuel, compared with energy obtained from the LWR. Dry reprocessing of LWR fuel followed by a recycle in CANDU provides an additional 50% energy.

Thorium-based cycles are an alternative option for the future and operate at near breeder efficiency. They need to be used with natural uranium or plutonium as driver fuel.

5.7.3 Gas Reactors

In the UK, Magnox and AGRs have achieved very high levels of performance. For AGRs, only a handful of failures have occurred in numbers of rods irradiated in excess of several million. Burn-ups have extended from 18 to 24 MWd kg^{-1} and beyond.

AGR stations have used different fuel designs, often designated Stages 1 and 2. The Stage 2 design has a single thick graphite sleeve with pins clad in multi-start ribbing in comparison to the original Stage 1 transverse rib design. The multi-start ribbed pins have improved heat transfer characteristics. For higher burn-ups discrete rings of gadolinia in stainless steel tubes have been added to the fuel elements.

5.7.4 Fast Reactors

Targets were set for the EFR design. Burn-up targets of 20% with refuelling at one-year intervals were an economic requirement. Fuel pin and sub-assembly designs were tested in fast reactors in the UK (PFR) and France (Phénix). The objective of the irradiations was to prove the fuel integrity with selected cladding materials for the design values of burn-up at fast neutron dose.

5.8. FUEL RESEARCH

Significant areas of research to support the present generation programme include safety and performance at all stages of the fuel cycle, reactor safety during plant operation, radioactive waste management, radiological protection, and other activities to benefit from ‘lessons learned’ in the past. There are active work programmes in all these areas. For example within the European Union there have been numerous activities funded by the European Commission Euratom Programme and corresponding counterpart national programmes (European Commission, 1994).

Regarding the fuel cycle, a key objective of the European Programme is to explore innovative approaches. Alternative fuel cycle concepts are being considered, primarily to address the problems of safeguarding long-lived radioisotopes. A particular example is partitioning and transmutation (P&T) which aims to provide a process of reducing the level of long-lived radioisotopes in high level waste. These methods rely on complex separation techniques and methods of transmutation that have yet to be developed. These techniques are in an early stage of development and are not yet prototyped at the industrial scale. The research programme is considered later in the book.

5.9. REPROCESSING

There are differences in national approaches in respect of once-through vs. a reprocessing and recycling policy. These approaches are linked with national policy on the management of natural resources, view on the relative radiological risk, domestic energy resources, security of supply and the relative economics, see for example, Bertel and Wilmer (2003).

From a sustainable energy perspective, recycling offers the option of better utilisation of resource and reduced radioactive waste. A MOX fuel cycle offers plutonium burning and reduction of radiotoxicity of spent fuel. In terms of public risk, an NEA study (OECD/NEA, 2000) concluded that the differences in public exposures between the fuel cycles were not significant.

The position adopted on the second and third issues depends on the country's requirement for autonomy.

The economics depend on the expected prices of uranium and fuel cycle costs and specific national conditions. The current position favours the once-through option, even with a significant growth in nuclear energy production.

5.10. SPENT FUEL MANAGEMENT

Waste management issues are discussed in more detail in the next chapter. Some countries have already put in place schemes for the disposal of high level waste in geological repositories; others have not yet committed to this approach.

There are additional safety issues associated with the storage of spent MOX fuel since MOX fuel generates more heat than UO_2. It may, therefore, be necessary to down-rate dry waste storage. A further point is that storage pools may require additional neutron poison to ensure adequate sub-criticality.

5.11. SUMMARY OF POSSIBLE FUTURE TRENDS

A good summary of future fuel cycle issues and reactor strategies over the next few decades is given in Meneley (1998). This report considers short-, medium- and long-term time frames extending out for the next few decades. Clearly the choice of reactor and fuel cycle are inextricably linked. For example, the most widely operating reactor type is likely to be thermal reactors burning mixed uranium and plutonium fuel. As discussed earlier, the fast reactor could be operated as a stand-alone technology or in combination with thermal reactors. There is then the possibility of the thorium fuel cycle.

The largest change is the introduction of MOX fuel in LWRs and HWRs. PWRs are already being loaded with up to 30% MOX fuel. Higher percentage MOX fuel loadings are being considered but further technical work is required to establish whether fission gas release at high burn-up is a concern. There is also the question of high burn-up fuel under accident conditions. The capability of multiple recycle is also not assured; it may be that MOX fuel is limited to two or three cycles. MOX fuels are feasible for up to 100% loading in HWRs.

Further development and proof testing of fuel elements, either of MOX or uranium fuel, will be necessary for fuels capable of utilisation to higher burn-up. This will mean higher fresh fuel enrichment. It is expected that there will be a continuous drive towards higher burn-up because of the improved economics, certainly for batch rods.

For HWRs, the life of fuel can be greatly increased by a small amount of enrichment. Natural uranium imposes an inherent limit on fuel life. This enrichment leads to more flexibility in design and fuel management. RU can also be used in HWR since the U-235 content of uranium remaining after plutonium extraction is about 0.9%. A sequential once-through cycle in two different reactor types is under construction called ' double-burning'. The idea is to use discharged fuel from the first cycle for the second cycle without re-enrichment. Another cycle is the 'DUPIC' cycle, which aims to reform LWR pellets into HWR pellets.

In the short term over the next 15–20 years there will be an opportunity to conduct small-scale fuel development experiments, before prototyping in large-scale experiments in the medium term. It is likely that uranium-based fuel will take precedence over thorium-based technologies but there is the possibility for more consideration to be given to the latter. In the longer term, it is possible that recycling will be a more routine practice. Either the FBR or accelerator breeding could be used to convert fertile material to fissile material in large quantities. Thorium would have the advantage over uranium of a very high conversion ratio.

Future work programmes could, therefore, focus on increasing the reactor conversion ratio resulting in higher burn-up for a given enrichment, and reducing the need for burnable poisons. This could be achieved either through a thorium cycle in thermal reactors or FBRs utilising metal uranium–plutonium fuel. Other research will target increased fuel burn-up, and reduction of reprocessing costs. Finally, on-line fuelling carries with it none of the disadvantages of periodic shut-down of batch fuelling. Flexibility is much increased and parasitic neutron absorption is reduced for fuelling at full power.

REFERENCES

Bertel, E. & Wilmer, P. (2003) Whither the nuclear fuel cycle. *Nucl. Energy*, **42** (3), 149–156.

Energy Visions 2030 for Finland, (2003) *VTT Energy*, Edita Prima Ltd, Helsinki.

European Commission: Nuclear Fission Safety, Reactor Safety, Waste Management and Radiation Protection, Work Programme, Edition 1994, Annex 1.

Groullier, J.P. *et al*. (2001) Different possible scenarios for plutonium recycling in PWRs. *Global Conference Proceedings, Paris, September 9–13, 2001*.

Hesketh, K. (2003) Advanced fuel designs for existing and future generations of reactors: driving factors from technical and economic points of view, FISA 2003: EU Research in Reactor Safety, Luxembourg, 10–13, 2003.

IAEA-TECDOC-984 (1997) Advances in heavy water reactor technology, *Proceedings of a Technical Committee Meeting held in Mumbai, India, 29 January–February 1996*.

IAEA-TECDOC-1122 (1998) Summary: fuel cycle options for light water reactors and heavy water reactors, 28 April–1 May 1998.

Ion, S.E. & Bonser, D.R. (1997) Fuel cycles of the future. *Nucl. Energy*, **36** (2), 127–130.

Leclercq, J. (1986) *The Nuclear Age*, Sodel.

MacDonald, P.E., Herring, J.S. & Neimark, D.C. (1998) Technical basis for the proposed high efficiency nuclear fuel programme, IAEA-TECDOC-1122, 28 April–1 May, 1998.

Malone, J.P., Varley, G. & Goldstein, L. (1998) Overview of economic and technical issues related to LWR MOX fuel usage, fuel cycle options for light water reactors and heavy water reactors, IAEA-TECDOC-1122.

Meneley, D.A. (1998) Future fuel cycle and reactor strategies, fuel cycle options for light water reactors and heavy water reactors, IAEA-TECDOC-1122.

Methnani, M. (2003) Status of high temperature gas-cooled reactor technology, IAEA HTGR Workshop ICTP Centre, Trieste, Italy, July 7–11, 2003.

OECD/NEA, (2000) *Radiological Impacts of Spent Nuclear Fuel Management options: A Comparative Study*, OECD, Paris.

Stanbridge, J.R. & Howl, D.A. (1992) Overview of nuclear fuel performance. *Nucl. Energy*, **31** (1), 21–23.

Tochihara, H., Komano, Y. & Ishida, M. (1998) The concept of a breeding PWR. *OECD/NEA ARWIF Conference Proceedings*, *Villigen, Switzerland, October 21–23, 1998*.

Youniou, G., Delpech, M., Girieud, R. & Guigon, B. (1998) Plutonium multi-recycling in a 100% MOX core with a high moderation ratio. *OECD/NEA ARWIF Conference Proceedings*, *Villigen, Switzerland, October 21–23, 1998*.

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Chapter 6

Waste Management and Decommissioning

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Chapter 6

Waste Management and Decommissioning

6.1. INTRODUCTION/OBJECTIVES

There are still major issues associated with the disposal of nuclear waste. There are bodies of opinion within the nuclear industry, regulators and many experts that believe solutions exist for all stages, but there is considerable public mistrust. This is fuelled since in many countries, there is no position on the final disposal strategy for long lived high-level waste, i.e. only temporary solutions are in place and there is no long-term policy. However, forward progress is happening in the US, Finland and Sweden where repositories are now being considered. Good progress has already been made towards the incarceration of low and intermediate waste in final long-term repositories.

The nuclear industry in common with all other industries has facilities that eventually come to the end of their productive life. Decommissioning of these facilities is then required, which involves the safe disposal of various hazardous materials. Such activities are carried out as a normal practice in an on-going nuclear energy programme and much experience has already been gained from a programme that has already been in operation for over 50 years. However, many present day reactors built in the 60s and 70s are now approaching the end of their design life and therefore decommissioning activities will increase over the next few years. This chapter also considers the key issues of decommissioning, including a review of different options that are being adopted and the impact on costs.

6.2. WASTE MANAGEMENT

6.2.1 Scale of the Problem

The quantity of radioactive waste produced from all sources is a very small fraction of the overall waste produced. In France for example, which has the highest fraction of its power generated by nuclear power, about 84,000 t of the 650,000,000 t of total waste produced annually is radioactive (Rosen, 1999) The latter figures include 200,000,000 t of hazardous industrial waste, yielding a percentage of radioactive waste in the hazardous waste of 0.015%. US figures are comparable.

The solid wastes produced from diverse energy sources for a 1000 MWe power plant are shown in Figure 6.1. A coal plant produces annually about 320,000 t of ash, containing about 400 t of hazardous heavy metals such as vanadium, mercury, and others. Additionally without abatement, a further 44,000 t of sulphur oxides and 22,000 t of

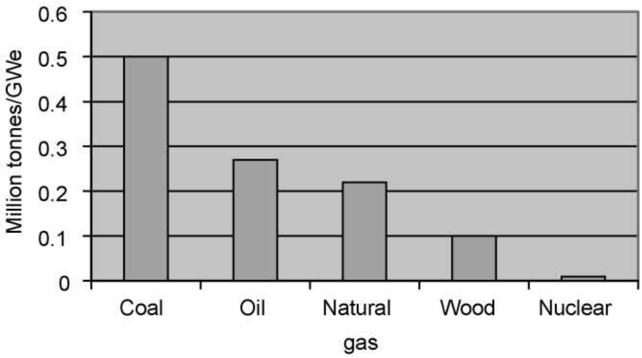

Figure 6.1. Waste from diverse energy sources produced annually. Source: Rosen (1999).

nitrous oxides go into the atmosphere and further waste is produced from mining and transportation. By comparison, a corresponding nuclear plant produces annually about 30 t of high-level waste (spent fuel) and about 800 t of intermediate- and low-level waste with virtually no release of noxious or greenhouse gases. Additionally, the waste quantities for fossil power plants are significantly increased by modern abatement techniques; e.g. sulphur abatement procedures for coal plants produce about 500,000 t of solid waste.

The management of radioactive waste is largely through confinement, since the quantities are extremely small. This contrasts the approach for large quantities of other toxic waste, which are dispersed in the environment to a level that is considered to be safe. Because of the large quantities involved, this is the only practical solution, yet clearly there may be safety concerns with this strategy.

Radioactive waste is typically characterised at three levels, low, medium and high. The levels of activity are categorised in different ways, but generally low level waste is deemed to be at a sufficiently low level of activity that shielding is not necessary apart from simple protective measures for handling. At intermediate level, shielding would be required; at high level, thick shielding and certainly remote handling facilities would be necessary.

For the purposes of waste disposal, the timescale of decay of the various isotopes will be an important factor, determining the time of confinement, and the facilities that are required for confinement.

Radioactive waste can come from many sources in the modern world. Most intermediate- and all high-level waste arises from civil nuclear power and military operations. In nuclear power activities, such waste arises from all stages of the fuel cycle; the significant waste problems arise from spent fuel and in waste from reprocessing operations.

Table 6.1 indicates the typical quantities and levels of waste arising from a 1000 MWe nuclear power plant.

Table 6.1. Quantities and sources of waste per annum from a 1000 MWe nuclear power plant

Waste category	Volume (m^3)	Sources
Low	200	Clothing, cleaning residues, machine components, filters
Intermediate	70	Contaminated equipment, reactor components
High	10 (2.5)	Spent fuel, concentrated liquid, (vitrified waste)

Rosen (1999).

In terms of worldwide production, the total volume of low-level waste is $\sim$100,000 m^3 per annum, compared with about 4000 m^3 per annum of high-level waste.

The present strategies for waste management depend on the relative levels of activity.

Low-level waste is usually stored in steel drums and disposed of in surface trenches above the local groundwater level. Since many of the isotopes in low-level waste have half-lives of only a few decades, the timescale for the waste to no longer pose a radiological hazard may be the order of only 100 years. The containment has to be sufficiently robust to resist corrosion and leaching of material for only a relatively short period.

Intermediate waste is encased in cement, inside steel drums. These are disposed of in relatively near surface repositories in a number of countries. Many repositories are already in operation and further facilities are expected in the future.

For high-level waste, an initial period is required to allow some decrease of its radioactivity and for residual heat to dissipate, before it is practical to consider long-term storage. Storage of spent fuel is usually under water initially at the site of production; in the longer term, dry storage may be possible. There is an intention in many countries to store high-level waste in deep underground repositories but as stated earlier this is not yet realised in most countries (Finland has recently given permission). To reduce volume, there is also the intention to vitrify high-level liquid waste to facilitate storage before disposal.

6.2.2 International Positions

The US has recently taken a decision to proceed with a spent fuel and high-level waste repository at Yucca Mountain (IAEA/NSR/2002, 2003; Figure 6.2). This is an important development, which has been met with considerable opposition and challenges to the supporting safety case. The repository will be under the control of the USDOE.

There has also been significant progress in Finland and Sweden. A good review of the important issues is given in Ryhanen (1996) together with a status commentary of the position in Finland, a leading country in developing long-term waste disposal strategies.

Some typical examples of waste disposal principles, identified in Ryhanen (1996) are the following. International recommendations exist for the various stages of waste

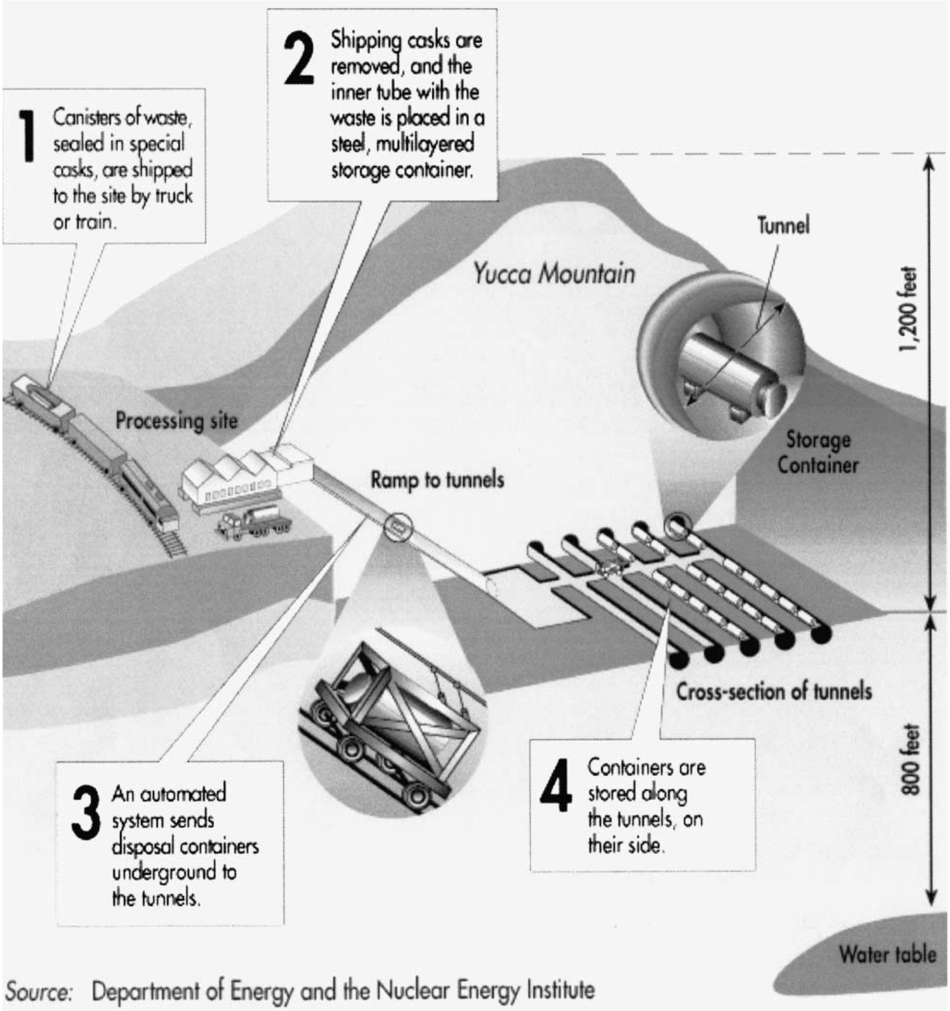

Figure 6.2. Yucca Mountain disposal facility. Source: http://www.nrc.gov.

management, and these are reflected in national legislation, albeit with some specific national modifications. Waste management facilities are licensed by national regulatory bodies. Relevant research, including experimental and theoretical R & D programmes, is conducted to confirm the technologies. An important issue concerns the financing over future years, e.g. one principle is that these costs should be recovered from ongoing electricity revenues, and set aside to guarantee future funds, but finding an appropriate model is an issue in many countries.

It is widely accepted that improved communication is needed to educate the public on the safety of the proposed technologies. It is not always clear what the public's concerns actually are. It is not always the safety issues that dominate the argument on long-term disposal. Could the presence of a waste disposal site impact on the commercial success of a region, e.g. by militating against the sale of its produce or through an adverse effect on tourism? Is the idea of final disposal less attractive than long-term temporary storage, the latter implying easier monitoring?

Many of the technical issues are complicated, covering wide ranging topics including organisational frameworks and responsibilities, technical details of the disposal, site selection criteria, licensing proceedings, etc. Therefore, an objective in communicating the technical issues to the general public is to make these as simple and straightforward as possible. In Ryhanen (1996), a number of simple observations are suggested. Is the risk of environmental pollution from spent fuel realistically perceived? For example, much of spent fuel is relatively insoluble, its radioactivity decreases with time, it does not radiate to the surface from an underground repository, etc.

Different groups of people need to be targeted in different ways. The most important groups to be targeted are the political decision makers both at national and local levels and the public. In Finland, the Posura Company's information programme ranges from press conferences, meetings with the municipalities, open houses to the public, exhibitions and lectures. Presentations are tailored to the particular audience and supported by relevant documented material. (Posura is a company that has been established by the Finnish Utilities TVO and IVO to specifically address the issue of the final disposal of spent nuclear fuel.)

It is clear that much understanding and progress has been made towards meeting the concerns of the problem of waste management and the management of spent fuel. However, it is also clear that much work is still required; this will need time and patience if the goal of finding an acceptable long-term solution is to be achieved.

Within Europe, the EC is likely to set a timetable in the near future for Member States to identify sites and to set up repositories for spent fuel and high-level waste disposal. The objective is to accelerate the various delays in decision making on the waste disposal issue that exist in some countries.

OECD/NEA is also encouraging countries to find long-term sustainable solutions to the waste problem (NEA Annual Report, 2002). It is aiming to facilitate improved technical and societal confidence in geological disposal in repositories. Activities in 2002 have included peer reviews of the Belgian and Swiss proposals, workshops on stakeholder involvement and technical reviews on the status of engineered barrier systems and ways in which geological science can be used to support repository safety cases.

The IAEA has recently reviewed the major issues and trends in radioactive waste management (IAEA/NSR/2002, 2003). This review reinforces the importance of the social and political aspects of radioactive waste policy. Issues relate to control of discharges and the availability of a retrieval option from repositories, particularly for spent fuel and high-level waste. A long-established principle is that waste should not impose an 'undue burden on future generations'. This is now being broadened towards the idea of an 'obligation'; that the current generation should avoid taking irreversible actions that may mean that certain necessary or desirable future options are not available to future generations.

6.2.3 Long-Term Disposal Research

Radioactive waste management, disposal and decommissioning are important areas of European research. Attention has focussed on evaluation of long-term disposal systems, including packaging policies and properties. Different aspects of waste retrieval are also under consideration. Underground facilities provide the best means of characterising potential disposal sites and for investigating different concepts of deep geological disposal. They are also required for data collection on the performance of the different barriers of protection. The performance of different types of rock ranging from clay, salt, marl or crystalline rock is being considered at possible sites. Within the EU, there are underground research facilities in the Asse salt mine in Germany and in the Hades facility in the Boom clay layer beneath Mol in Belgium. In France, some experimental activities are on-going in existing mines, e.g. the Amélie mine; sites for other underground laboratories are being considered.

Research tasks cover the testing of different methods of disposal, the methods of backfitting and scaling of repositories. They have also included the investigation of the long-term behaviour of components and groundwater flow and the migration of radionuclides.

An important objective of the research is to gain an improved understanding of the essential phenomena. The main requirement is to understand the release of radionuclides from the waste packages and their migration through the repository barriers to the environment. Characterisation of the different levels of waste in the waste volume is useful, to reduce the volume of highly active waste for disposal in deep underground repositories. Research is carried out into characterising the different waste forms and matrices (cement as a containment and barrier material, spent fuel itself and glass matrices of vitrified material). The quality control of nuclear waste packages and waste forms is being promoted to facilitate standardisation of checking methods, a common understanding of R & D requirements and unification of test methods, etc.

The mechanical and chemical stability of the engineered barriers and the surrounding host rock is influenced by groundwater movements, thermal energy transfer effects at the interface and beyond and the radionuclide transport. It is important to understand any long-term degradation of these engineered barriers. The generation of gas can occur due to a number of processes affected by the nature of the waste, waste package, buffer and backfill materials and the nearby host rock. This could result in the build-up of pressure and possible structural problems in the repository.

Radionuclide migration research focuses on the thermodynamics of the solid-liquid phases' equilibria and complexes with organic materials. These include groundwater colloid generation and transport, transport and retardation processes through porous and fractured rock systems and chemical thermodynamic and kinetic processes associated with radionuclide transport through the engineered barriers.

Studies have been performed on natural geological sites and have provided qualitative and some quantitative data on geochemical aspects (e.g. container corrosion, waste form degradation, radionuclide solubility and transport processes).

Palaehydeogeological studies also provide information on site evolution over geological time scales. Information of ancient flow patterns can provide understanding of past rates of uplift, erosion and of other, e.g. climatically induced changes of groundwater behaviour.

In addition to technical research, there are EU research programmes to enhance public understanding of the impact of waste disposal and to establish better methods of achieving public confidence and trust (European Commission, 2001). The objectives include gaining an understanding of the origins of public mistrust, evaluation of better means of communication and evaluation of decision making at different levels (e.g. local, national and international).

The decommissioning of nuclear installations is the final chapter in closing the nuclear fuel cycle. Research programmes are in place to develop innovative dismantling techniques to collect technical performance data, including data on specific wastes and doses arising from decommissioning.

6.3. WASTE MANAGEMENT POLICY

National regulators govern the waste management programmes in their countries. Nevertheless, there is a considerable harmonisation of policy and principles in regard to waste disposal safety. The UK approach is considered by way of example (Cmnd 2919, 1995). There are also however differences, particularly in regard to the high-level waste disposal issue, already discussed earlier.

In the UK, the same legislative framework exists for waste management and decommissioning, as exists for operating nuclear power plant. Activities are governed by the Health and Safety at Work Act, 1974 and the associated statutory provisions of the Nuclear Installations Act, 1965. More details are given in Chapter 8.

The UK national policy was reviewed in the 1995 White Paper; the conclusions are given in Cmnd 2919 (1995). The UK Health and Safety Executive (HSE) has defined 10 policy issues. These are summarised in Table 6.2.

Table 6.2. HSE policy issues for radioactive waste management

Issue	Requirements
Strategic planning	Licensees must develop programmes within an appropriate timescale
Site-specific waste strategies	Licensees must provide for the management of all radioactive waste on site
Continuity of radioactive waste management responsibilities throughout a licensee's period of responsibility	HSE must manage radioactive waste on site through to the end of their period of responsibility under NIA65
Generation of waste	Waste is not unnecessarily created and the generation and accumulation of waste should follow ALARP
Balance of risks to workers, the public and the environment	The total detriment should follow ALARP
Segregation and characterisation of wastes	Where practical and cost effective, waste should be segregated to facilitate the overall safe management of conditioning, storage, retrieval and subsequent disposal
Disposal of radioactive waste	Disposals in accordance with RSA93
Safe storage of radioactive waste	Where practical and cost effective, it should be stored in a passively safe form and in a manner to facilitate final disposal
Retrieval or transfer of stored waste	HSE expects that new waste storage facilities should be designed with retrieval and transfer in mind
Project use of storage facilities	Existing waste forms and waste storage facilities should be reviewed through an appropriate maintenance and surveillance programme

Bacon (1997).

A key requirement is the need for strategic planning. Where disposal routes exist, the general principle is to move towards long-term storage with the waste in a passive safe form, rather than an approach that requires frequent monitoring.

6.4. GENERAL FACTORS IN DECOMMISSIONING

Decommissioning and waste management form part of the deeper problems that nuclear power has to face in today's society. It is argued in Wilkie (1996) that the issues are neither technical nor just financial. Although technical issues remain, much progress has been made, and the remaining problems may be solved with sufficient financial support. This may be costly from a financial perspective, since many of the nuclear facilities that require to be decommissioned were not designed to take this into account. Nevertheless, this is not the whole problem. The problem is that the liabilities of decommissioning, e.g. spent fuel and other radioactive wastes remain hazardous over very long time scales, for hundreds of years. To deal with this problem requires stable national frameworks to sustain an appropriate nuclear industry with the required technical skills for a similar period.

The nuclear electricity generating industry involves the availability of various facilities. Facilities involved in a thermal reactor fuel cycle of the type required to support the UK reactor programme are described in Gordelier (1997). For such a cycle, the stages involve the mining of uranium ore, followed by an appropriate chemical treatment plant to produce the required uranium fuel. Following this, fuel is fabricated in a fabrication plant ready for loading into the reactor. If the fuel is to be recycled, it would be sent to a reprocessing factory where the uranium would be recovered for future use. Recovered plutonium might be for the production of MOX fuel or for utilisation in a fast reactor fuel cycle, perhaps in the future. Residual radioactive material from the fuel would be sent for appropriate storage, treatment and waste disposal. If the fuel is not reprocessed, then it would be treated as waste and sent for high-level waste storage. There are therefore many and diverse facilities associated with the fuel cycle that at some stage will come forward for decommissioning.

Different facilities pose different problems. For example, chemical treatment and fabrication plants that handle first pass fuel are relatively easy to decommission since they only handle low radioactivity materials. On the other hand, facilities that handle recycled uranium and particularly plutonium, e.g. reprocessing plants, pose a much greater challenge. The decommissioning of the power reactors themselves is also a major challenge.

There are various issues that need to be considered in the decommissioning of nuclear facilities. Many of these relate to the timing of decommissioning and dismantling and the factors that determine strategy. Some general principles are set out later, see for example Twidale (1999).

Clearly the safety of radioactive and other hazardous materials is of paramount importance. The safety of the facility will have been assured by its safety case for operation. Operations for its decommissioning phase will need to be covered in an on-going safety case consistent with the relevant national government legislation. In the UK, the policy for decommissioning is to systematically reduce the hazards until the site can be freed from licensing constraints. This is set down in the UK Government's waste management policy (Bolton, 1996).

After shut-down, defuelling and all other clean-up operations need to be managed to minimise any risk to the general public, the workforce and the environment. The defuelling process removes a high percentage ~99.9% of the radioactive inventory.

In general, decommissioning should commence as soon after cessation of operations as is reasonably practicable. In particular, post-operations clean-out (POCO) should be carried out early in the decommissioning process to reduce any radioactive contamination within the plant.

The management of waste must be consistent with long-term disposal plans and no action should be taken that might prevent these plans being carried out. The quantities of waste should be minimised, e.g. waste and fuel handing operations should avoid double-handling operations if possible.

The timing of the above operations will be dependent on the existence of facilities to retrieve waste, and waste disposal routes need to become available. This could impact on the timescale for dismantling the plant, consistent with the maintenance of safety. Processing plants need to be in place together with the long-term storage facilities.

6.5. DECOMMISSIONING STRATEGY

This section considers various decommissioning strategies and the options available.

Generally, the ultimate objective in decommissioning is to return the site to a state whereby it can be used without restriction (de la Ferte, 1996). The IAEA have defined three stages of operation, see Table 6.3. The timescale for carrying out these activities will depend on the decommissioning strategy. Work may proceed from one stage to the other relatively quickly or may take place over many decades, perhaps over as many as a 100 years.

Table 6.3. Stages of decommissioning

Stages	Potential periods of activity
1. Removal of nuclear fuel removes 99.9% of the radioactivity	Rapid progress from one stage to the next; or activities carried out over a prolonged period, 100 years or more
2. Dismantling of structures, e.g. other than the reactor itself and its surrounding biological shield	Possible hold points
3. Total dismantling, removal of all materials with radioactivity exceeding natural background	Complete all stages; remain at Stages 1 or 2 for a relatively long period; or proceed directly from Stage 1–3

de la Ferte (1996).

Table 6.4. Decommissioning options

Options	Features
1. Safe enclosure	Following defuelling as rapidly as possible enclose the active inventory without immediate dismantling
2. Safe enclosure together with partial dismantling	Similar to Option 1 except with partial dismantling and storage of components on site for total dismantling later
3. Immediate total dismantling	Total dismantling with removal off site of all waste materials

Eßmann (1990).

On this basis, a number of options are available to the operator. Table 6.4 summarises these options. Option 1 or the ‘safe enclosure’ option leaves the plant essentially unchanged after the completion of Stage 1. Once the entire operating medium, e.g. the fuel has been removed, all the nuclear plant equipment is sealed. The objective of safe enclosure is to enclose any remaining activity as soon as possible without immediate dismantling and then when this has been achieved to wait for the inventory radioactivity to reduce by natural radioactive decay.

Option 2 or partly dismantling with safe enclosure involves placing certain active components of the plant obtained by dismantling along with other plant components in a safe store. The principle of this approach in terms of environment protection is similar to that of Option 1. Total dismantling will be completed at a later date, once the inventory has reduced sufficiently by natural decay.

Option 3 is based on the premise of total dismantling. Here all active and inactive waste materials are removed from site directly after the end of operational life.

There are various important technical, safety and economic issues that need to be addressed in all decommissioning programmes. These are summarised in Table 6.5.

Table 6.5. Important issues to be considered in decommissioning

Technical aspects – structural integrity issues, inventory management and volume of material, degree of automation, remote handling requirements, decontamination arrangements, health physics and available dose minimisation techniques, material re-usage following decommissioning
Decommissioning policy – regulator requirements, licensee decommissioning strategy and workplan, timing of operations
Safety and environment – control of hazardous releases during decommissioning operations, waste treatment, temporary or permanent storage, repository storage
Radiological issues – adherence to ALARA principle for personnel exposure, advantage in delay in plant dismantling (safe enclosure)
Public relations – management of waste disposal concerns
Economics – relative benefits/disadvantages of ‘safe enclosure’ versus ‘immediate dismantling’

Eßmann (1990).

6.6. TECHNICAL ISSUES

The technical aspects of various activities that need to be considered in the planning of plant decommissioning are discussed in this section. The resolution of these issues will generally be site dependent and depend on the infrastructure for decommissioning that already exists, both at the national and local level.

Assurance of structural integrity fidelity and effective management of radioactive inventory are key pre-requisites towards ensuring the safe and efficient management of decommissioning operations.

6.6.1 Structural Integrity

By virtue of the plant operating licence, ageing plant structures must have sufficient integrity to meet all safety requirements during the latter stages of plant life. However, the impact on the structures from decommissioning operations must also be assessed prior to decommissioning. For example, in the case of the containment, there may be a need to erect another structure to meet the required containment safety function.

In the decommissioning of ageing plants, structural integrity of vital components cannot be expected to meet the present day standards. In this case, the adequacy of the components will need to be considered against ALARP principles.

6.6.2 Inventory Management

The condition of inventory will need to be managed to meet radiological, environmental and possibly other safety concerns.

For example, there may be chemical corrosion processes that affect the handling of fuel downstream in the disposal route. The timescale of these processes could impact the timing of certain operations depending on whether a corroded or an uncorroded state of the inventory is easier to manage. The gaseous chemical products of reaction may also be a concern, e.g. in Magnox plants, the Magnox/water reaction produces hydrogen (Twidale, 1999).

It may be possible to dilute liquid inventories as a means of reducing the specific radioactivity of the liquid. This could provide significant benefits in dose management of the work force. Further, by appropriate chemical treatment, it may be possible to reduce the impact on the environment.

Repackaging of the inventory into a safe form to meet all the necessary safety requirements is likely to be necessary. Interim storage is likely to be adopted in most countries where the approach for long-term storage, e.g. in a repository, has yet to be agreed.

6.7. DECOMMISSIONING POLICY

Decommissioning safety risk is primarily associated with risks associated with public health and safety and the risks associated with waste management (de la Ferte, 1996). All OECD countries with nuclear programmes have in place decommissioning regulations, either as part of their general legal infrastructures for nuclear plant licensing or specifically for decommissioning. The IAEA have also set down the general principles to be followed, and defined the respective responsibilities for regulator and operator.

National licensing procedures define whether the operator or public authorities are empowered to decide on shut down and decommissioning of facilities. There are some differences between countries in terms of responsibilities. In the UK and Germany, the responsibilities for the shut down and decommissioning of facilities lies solely with the operator under normal circumstances (Willby, 1996). In other countries such as France, the operator has less independence. There have also been instances where governments have taken a political decision to shut down plant as in the moratoria imposed by Italy and Sweden. The body that has the responsibility for decommissioning operations is also different in different countries, in Canada for example, it is the operator; in Belgium and Spain there is a specialised public agency responsible for radioactive waste management.

In the UK, the HSE has set down policy issues and broad requirements on the licensee (Bacon, 1997). These cover requirements on the licensee in regard to defining strategic plans, work plans, and schedules and priorities for the progressive reduction of hazards (Walkden and Taylor, 2002). These requirements are summarised in Table 6.6.

Table 6.6. HSE policy issues for decommissioning

Issue	Requirements
Strategic planning	Licensee expected to produce a decommissioning strategy for their plants and sites
Site or plant specific decommissioning programme	Licensees are required to produce programmes and arrangements for decommissioning
Timing of decommissioning	Licensee required to commence decommissioning at an agreed time with timing of specific projects reviewed periodically
Priorities	Systematic and progressive reduction of the hazards
Completion	HSE will regulate the safety of activities until it can advise that there is no further danger from ionising radiation

Bacon (1997).

6.8. SAFETY AND ENVIRONMENT

EC legislation exists to control the environmental impact of decommissioning activities (Nash and Woollam, 2002; Statutory Instrument No. 2892, 1999). This requires the provision by the operator of an environmental statement to support decommissioning activities. UK regulations have a similar requirement.

In general, operations must be managed to ensure that any hazardous releases are prevented as far as is reasonably practicable. Any plant conditions that have a potential for uncontrolled releases must be dealt with as soon as possible. Any discharges that do occur must be within agreed limits. If a potential to exceed these limits is recognised, e.g. prior to defuelling, then early decommissioning may be necessary (Twidale, 1999), or authorisations may have to be renegotiated.

6.9. RADIOLOGICAL ISSUES

The strategy in most decommissioning activities is to reduce the radiological hazard in a systematic way, until the delicensing condition for the site is reached. As noted earlier, after operations have ceased in a reactor, the removal of fuel reduces the hazard by a significant degree. After that the POCO will result in a further reduction in level.

In general, radioactive decay will result in a reduction of radioactivity and deferral of operations may be of benefit. However, there may be an issue if radioactive daughter chains exist producing isotopes that present greater problems than with the parent isotope. For example (Twidale, 1999), the ^{241}Pu isotope primarily emits Beta radiation but it has ^{241}Am as a gamma emitter. Furthermore, the parent isotope has a half-life of 12 years but the daughter has a half-life of 432 years.

The activation of the construction materials of the reactor and the presence of gamma-emitting isotopes are a significant problem in decommissioning. The areas concerned are the core internals, the biological shielding and the pressure vessel. The most problematic isotopes are ^{60}Co, ^{108}Ag and ^{94}Nb, which have half-lives of 5.27, 418 and 20,000 years, respectively. It is therefore possible to achieve reductions in activity from, e.g. ^{60}Co after a timescale of several decades, but the other problematical radioisotopes will remain.

The quantity of activated components varies considerably with the type of reactor and the size of the vessel. A large PWR has a reactor vessel of diameter 4.5 m and total weight 600 Te compared with a Magnox reactor that has a vessel of about 20 m of weight 5000 Te. The PWR vessel can be moved as a single item but this would not be possible for a Magnox reactor (Twidale, 1999).

6.10. PUBLIC RELATIONS

The issue of public concern surrounding decommissioning is largely centred on the concerns of safe waste disposal (de la Ferte, 1996).

6.11. ECONOMICS

For many of the facilities that are currently being decommissioned, little attention was given to decommissioning in their design (Review of Radioactive Waste Management Policy, 1995). This has resulted in an increase in costs in some cases. In the UK for example, the regulator now requires that consideration be given to decommissioning in the design of a plant. This is in regard to a number of factors, construction techniques, choice of materials, the provision of suitable access and the availability of adequate waste storage facilities.

The costs of decommissioning for different reactor types and different countries were considered in an IAEA review of selected cost drivers for decisions on the continued operation of the older nuclear reactors (IAEA-TECDOC-1084, 1999). This review covered pressurised water reactors (PWR and VVER), BWRs, HWRs, light water cooled, graphite moderated reactors (LWGW or RBMK type) and gas reactors (GCR and AGR).

Two categories of decommissioning costs are considered. The first category (Stage 1 and/or Stage 2 decommissioning followed eventually by Stage 3) is decommissioning with long-term storage. This takes advantage of the natural decay of the radioactive isotopes, which makes dismantling operations at a later time much easier. The second category is the decommissioning approach with immediate dismantling of the plant up to the 'green-field' (non-restricted use) or 'grey-field' (somewhat restricted use) condition (Stage 3 decommissioning). This structuring of the decommissioning stages is based on the well-established IAEA terminology.

It is noted that decommissioning practices differ substantially from country to country and this affects any comparable cost estimates. For example in some countries, the cost of fuel unloading is included as a standard part of decommissioning costs. In most countries, it is not. There is not necessarily a consistent practice within a particular country. The study in IAEA-TECDOC-1084 (1999) aimed to focus on total costs and made no attempt to consider the relative importance of various cost components.

Section 6.11.1 summarises the estimated costs of decommissioning after storage, for the principal types of reactor in operation at the present time. Data are taken from IAEA-TECDOC-1084 (1999). It should be noted that the costs considered were total costs excluding discounting. It should further be recognised that not all costs in the data were normalised to exactly the same time period.

Table 6.7. Decommissioning after storage costs

Reactor	Power range (MWe)	Cost ($US per kWe)	Comment lower/higher range of cost
PWR	500–1300	200–700	Germany, US (lower), Netherlands (higher)
BWR	160–1300	150–600	Finland, US (lower), Germany (higher)
VVER	440	120–1400	Russia (lower), Germany (higher)
HWR	540–1300	100–380	All Canada
RBMK	1000	180–600	Russia (lower and higher)
GCR and AGR	200–600	1000–3000	UK (lower and higher)

IAEA-TECDOC-1084 (1999).

6.11.1 Decommissioning After Storage

In general, the IAEA Study (IAEA-TECDOC-1084, 1999) found that the variations and uncertainties found in the data (costs levelised to 1997) for decommissioning after long-term storage (Table 6.7) had similarities with the data for decommissioning immediately (Table 6.8). In particular, this correspondence related to variations from country to country and also from specific case to case. Also not surprisingly, decommissioning costs were sensitive to national labour resource estimates.

6.11.1.1 PWRs. Estimates were provided for reactors from Belgium, Germany, Japan, Korea, Netherlands and the US for units in the range 500–1300 MWe. It was found that costs ranged between 200 and 700$US per kWe, and for the capacity range considered there were no economies of scale. It was found that in most cases the total over-night costs for decommissioning with long-term storage were higher than for immediate decommissioning, considered later. However, it is recognised that the net present value would normally be on the contrary, dependent on the decommissioning schedule and the assumed discount rate.

6.11.1.2 BWRs. Estimates were provided and shown for Finland, Germany, Italy, Japan, Netherlands and the US for reactors in the range 160–1300 MWe. Decommissioning costs were found to be in the range 150–600$US per kWe (a small BWR-60 plant in

Table 6.8. Immediate dismantling decommissioning costs

Reactor	Power range (MWe)	Cost ($US) per kWe	Comment lower/higher range of cost
PWR	500–1400	150–700	Finland, Sweden, US (lower), Netherlands (higher)
BWR	470–1300	170–650	Germany (higher), Finland, Sweden, US (lower)
VVER	440	120–1240	Russia (lower), Germany (higher)
HWR	200–1300	130–310	India (lower), Korea (higher)
RBMK	1000–1500	50–100	Russia (lower), Lithuania (higher)

IAEA-TECDOC-1084 (1999).

the Netherlands was also analysed; it was found that scaling effects did exist for the smaller capacity plant). In addition, the relative costs between decommissioning after long-term storage and immediate decommissioning were similar to those for PWRs. For example, long-term storage undiscounted costs were again higher than immediate decommissioning but as noted earlier, the situation would be different if discounted costs are taken into account.

6.11.1.3 VVERs. Data were available from Bulgaria, Czech Republic, Germany, Russia and Slovakia. Reactors considered were the VVER 440 MWe plants 440/230 and 440/213. The costs were found to differ widely from 120 to 130$US per kWe in the Russian Federation up to 1400$US per kWe in Germany. Much of this difference is reflected in labour rate costs. As for PWRs and BWRs, decommissioning with long-term storage is more expensive than immediate dismantling (undiscounted costs). The differences were relatively small for Slovakia.

6.11.1.4 HWRs. Decommissioning costs from three available assessments of Canadian units were estimated. It was found that there was a substantial difference between units of similar capacity, largely reflecting the situation that the cost estimates were made at different times. For some Canadian plants, decommissioning with long-term storage was found to be cheaper than decommissioning with immediate dismantling even in undiscounted costs.

6.11.1.5 LWGRs (RBMK). Estimates were available from the Ukraine and Russia. Due to the large amount of graphite in the core, decommissioning with long-term storage is a more feasible option for LWGRs than decommissioning with immediate dismantling. Most assessments for long-term storage are again higher (undiscounted) than those for immediate dismantling.

6.11.1.6 GCR and AGRs. Gas-cooled reactor data were supplied from the UK and for an old reactor in Spain, results are shown for the range 200–660 MWe. Due to technical design reasons, decommissioning with long-term storage is preferable to immediate dismantling. This is because there are some operations that can be carried out manually that are not possible for PWRs and BWRs. The costs for GCRs are higher than for other reactor types 1000–3000$US per kWe for the above capacity range. Part of the reason is not only due to the smaller size of GCR units but also there are larger volumes of radioactive waste that need to be processed. There are also increased man-power requirements.

6.11.2 Immediate Dismantling

Immediate dismantling decommissioning costs are summarised in Section 6.11.2, for the major reactor types of interest (IAEA-TECDOC-1084, 1999).

6.11.2.1 PWRs. Data were collected from a number of countries including Belgium, France, Korea, Netherlands, Sweden, UK and the US. A range of plants was considered covering the range 500–1400 MWe. The costs spanned between 150 and 700$US per kWe, reflecting large deviations in the key decommissioning parameters across the countries considered. These related particularly to differences in labour requirements, on the amount of decommissioning wastes and the duration of decommissioning activities. In general, it was found that the effect on reactor scale was small compared with differences between countries and differences between the estimates for the same reactor made at different times.

6.11.2.2 BWRs. Estimates were considered for Finland, Germany, Japan, Netherlands, Sweden and the US, covering reactor units in the range 470–1300 MWe. Decommissioning costs were found to be in the range 170–650$US per kWe, i.e. similar to those for the PWRs (the effects of scale, however, were more visible than for BWRs), but again these were small compared with cross country variations of estimates with time.

6.11.2.3 VVERs. VVER plants have certain design differences from PWRs which impact on decommissioning costs, e.g. there is a high share of common systems and components in twin units.

In IAEA-TECDOC-1084 (1999), costs were presented for Bulgaria, Finland, Germany, Russia and for Slovakia for largely 440 MWe units of the 230 and 213 specification.

In general, the costs for VVERs were similar to those for PWRs and BWRs, except in Germany and Russia. Costs were higher in Germany and lower in Russia. These differences were not quantified but differences in labour rates and also in labour requirements were contributing factors.

6.11.2.4 HWRs. Data were available from Canada, Korea and India covering plants in the range 200–1300 MWe. In general, costs for HWRs are of the same order as for PWRs, BWRs and VVERs, but the cost variation from case to case appears less. However, the sample of plants considered was smaller. The costs for Korea were higher than Canada and India.

6.11.2.5 LWGRs (RBMK). Estimates were considered from Lithuania (1500 MWe Ignalina NPP) and Russia (1000 MWe plants). In general, assessments for LWGRs are lower than for other types, almost certainly reflecting low labour rates in these countries. The estimates were higher in Russia than in Lithuania; however, in the Russian data the costs of handling irradiated graphite were not included. In the Lithuanian data, these costs were taken into account.

Practically all the costs above were derived on the assumption of planned decommissioning. There may be cases when decommissioning is required urgently. This might be due to economic, safety, political or social reasons. In such cases, additional financial losses may be incurred (IAEA-TECDOC-1084, 1999).

REFERENCES

Bacon, M. (1997) HSE policy on decommissioning and radioactive waste management at licensed nuclear sites, *Nucl. Energy*, **36** (1), 13–17.

Bolton, C. (1996) Plant life extension for early nuclear power plants, *Nucl. Energy*, **35** (1), 53–57.

Cmnd 2919, Review of radioactive waste management policy, Final conclusions, HMSO, London, 1995.

de la Ferte, J. (1996) Decommissioning nuclear installations: business as usual or playing with the devil? *Nucl. Energy*, **35** (5), 339–344.

Eßmann, J. (1990) Operators' view of key issues confronting nuclear power plant decommissioning. *Proceedings of the International Conference on Decommissioning of Nuclear Installations, Brussels, Belgium, 24–27 October 1989*, EUR 12690, Elsevier, Amsterdam.

European Commission: Research and Training Programme in the Field of Nuclear Energy (1998–2002), Work Programme, Revision August 2001.

Gordelier, S.C. (1997) Decommissioning update: the view at the end of the line, *Nucl. Energy*, **36** (3), 185–195.

http://www.nrc.gov.

IAEA-TECDOC-1084 (1999) Review of selected cost drivers for decisions on continued operation of older nuclear reactors, May 1999.

IAEA/NSR/2002 (2003) Nuclear safety review for the year 2002, International Atomic Energy Agency, August 2003.

Nash, R.S. & Woollam, P.B. (2002) The potential benefits and drawbacks of deferring the decommissioning and dismantling of nuclear facilities, *Nucl. Energy*, **41** (5), 325–331.

NEA Annual Report (2002) Radioactive waste management.

Review of radioactive waste management policy, final conclusions, HMSO, London, July 1995.

Rosen, M. (1999) Managing radioactive waste: issues and misunderstandings, *Nucl. Energy*, **38** (4), 249–258.

Ryhanen, V. (1996) Nuclear waste disposal – how to meet the concerns of the public. *Nucl. Energy*, **35** (5), 349–351.

Statutory Instrument (1999) No. 2892, European Communities, Nuclear Reactors (Environmental Impact Assessment for Decommissioning) Regulations, 1999.

Twidale, D.W. (1999) Life extension of magnox nuclear power stations, *Nucl. Energy*, **38** (2), 107–112.

Walkden, P. & Taylor, D. (2002) Reactor decommissioning in a deregulated market. *Nucl. Energy*, **41** (3), 219–222.

Wilkie, T. (1996) An outsider's view of decommissioning. *Nucl. Energy*, **35** (5), 345–347.

Willby, C.R. (1996) Regulatory issues in decommissioning. *Nucl. Energy*, **35** (3), 187–191.

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Chapter 7
Advanced Reactor Design

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Chapter 7
Advanced Reactor Design

7.1. INTRODUCTION/OBJECTIVES

This chapter describes advanced reactor design requirements and the status of international activities. The assumption is made that nuclear power will continue to provide a reliable and sustainable energy source, while complementing that produced from other technologies, e.g. fossil fuel, renewables, etc. It discusses general design objectives, primarily from a utility and vendor requirements perspective. Advanced reactors are often classified into two categories namely, 'evolutionary' and 'innovative'. In this context 'evolutionary reactor' refers to the class of reactors with relatively small modifications from existing designs. By contrast, 'innovative reactors' incorporate substantially new designs, which would require significant investment to develop. Potential regulator requirements for advanced plants are considered in the next chapter.

The primary raison d'être behind the design of current generation plants was that they should be able to provide a reliable and safe base-load electricity supply. The same requirement holds true today but with increased emphasis on economic viability, increased safety characteristics and improved public acceptance. These considerations are paramount in advanced reactor design specification. The main focus of this chapter is to describe international developments in design philosophy in advanced evolutionary reactors. The characteristics of a number of more innovative advanced reactor designs that are being proposed, including design requirements, are considered in more detail in the subsequent chapters.

7.2. INCENTIVES AND JUSTIFICATION

The main arguments for the continuation of nuclear power have already been discussed in earlier chapters, i.e. it offers a carbon-free energy source of energy via an established and proven technology. New plants are now needed for electricity generation to replace the plants that came on stream in the 1970s and which are now reaching the end of life. If new build programmes go forward, the issue is what plants to build? The drive for most new advanced evolutionary reactor designs is to achieve higher performance and safety by virtue of the design, rather than by simply improved operation which is often the only viable option available in the case for present generation plant. Satisfying these requirements is crucial to meet the increasing competition from natural gas electricity generators, coupled with an increasing trend of de-regulation. Broad design objectives are considered in Section 7.3.

7.3. DESIGN OBJECTIVES

Since the early days of nuclear power, a large number of different design concepts have been considered. These have now been licensed, built and operated successfully but not without significant effort and investment. Now that these systems are proven, the tendency has been to focus on evolutionary designs (IAEA-TECDOC-1117, 1999; Juhn, 1999), largely arising from the need to move forward cautiously. Regulators tend to adopt a conservative line in licensing new developments. Utilities seek to reduce risk by staying close to proven technologies. De-regulation of the industry and reducing investment from governments have resulted in the reluctance of building and financing new prototypes.

Evolutionary water-cooled reactors benefit from the wide range of experience that has been accumulated from operation of present generation plant. This experience has been embodied into the Electric Power Research Institute (EPRI) User Requirements Documents (URDs) and the European Utility Requirements (EURs) and other utility design guidelines. Such international experience has been disseminated through the activities of WANO and the IAEA.

Examples of improved performance that can be established during the design phase include IAEA-TECDOC-1117 (1999), some particular goals, e.g. short outages, overall simplicity of design and on-line maintenance. These can be achieved by improved man–machine interfaces, better computers and IT, more plant and component standardisation and better operator qualification and training. Better availability can be achieved by increased design margins, enabling greater robustness against reactor trip. This can also help in extending plant lifetimes, which are often limited by eroding margins in present day plant.

It should be possible to meet increasingly stringent safety objectives by incorporating new design features. The readiness of these features to improve accident prevention and mitigation has been tested.

Economic competitiveness is important for advanced plants in the same way as for present generation plants. Simplicity of design is a key factor. Further, designs must be substantially completed prior to start of production. This helps to avoid hold-ups in regulatory requirements, long construction delays and facilitates the operations' management. The increasing de-regulation of the electricity markets is a continuing drive to ensure designers strive for simpler designs, without compromising safety.

The financing of new plants is likely to require special conditions to minimise risk and, therefore, keep financing rates to a minimum. Conditions to be considered are a favourable national policy regarding nuclear power, economic competitiveness, feasibility of the project, adequate revenues, e.g. from long-term purchase agreements and no open-ended liabilities.

Finally public acceptance of new designs will need to be gained. This may be achieved by education and demonstration that severe accidents of the past would be eliminated through new design technologies.

7.4. UTILITY REQUIREMENTS

Utility requirements' documents have been produced which aim to provide direction to designers by taking advantage of experience from current plants. The aim is to reduce costs and uncertainties of licensing by demonstration of a sound technical basis for advanced designs.

7.4.1 EPRI Utility Requirements (UR)

EPRI, in collaboration with USDOE, have developed a set of requirements to establish the technical basis for the design of advanced light water reactors (ALWRs) (IAEA-TECDOC-968, 1997). A first objective is to establish a basis for licensing future LWRs, including the resolution of outstanding severe accident issues, and to gain agreement with the USNRC. Secondly, there is an objective to provide a standardised plant design with vendor certification. Thirdly, there is an intention to provide a set of technical requirements, to minimise the risks to investors in completing and operating the first ALWR.

The EPRI Utility Requirement Document (URD) covers top-level programme policy statements and detailed requirements for specific ALWR designs. It includes large evolutionary systems with improved active safety systems and also passive system designs including natural circulation, gravity-driven refill and stored energy as essential safety functions. Both passive PWR and BWR systems are included.

The document was first published in 1990 and has been used in the development of several new LWR designs. It has been developed by the US utilities to reflect the procedures' rules, regulations, codes and standards of the US. However, there have also been contributions from European utilities, which have developed their own set of standards, as discussed below.

7.4.2 European Utility Requirements (EUR)

The EUR were initiated in 1992 by a group of major European utilities from Belgium, France, Germany, Italy, Spain and the Netherlands. The group was later expanded in 1996, to include Finland and Sweden (IAEA-TECDOC-968, 1997). The initial objective was to develop a set of common safety requirements to be agreed eventually with the regulators. Later the intention was to extend the scope towards the development of standardised designs that would be licensable in the various countries in Europe. The intention was to facilitate the movement of nuclear industry in the European Union towards the open common market policy, while recognising the role of the independent national regulating authorities.

The EUR Document (EURD) covers the principal policies and high-level requirements. It also includes generic requirements for nuclear utilities that are not specific to a particular design. It does, however, also include some requirements for specific designs that are of interest to the countries that are participating in the initiative.

The EURD has been used in the development of the EPR design by the French and German companies, Framatome and Siemens, respectively. It has also been used for the design and development of the European Passive PWR (EPP) and the European Simplified BWR (ESBWR).

7.4.3 Japanese Utility Requirements (JUR)

The Japanese standardisation programme was a collaborative effort between the government and industry, led by the Ministry of International Trade and Industry (MITI) (IAEA-TECDOC-968, 1997). It started as early as the mid-1970s with the objective of standardising LWR designs on the operating plants of the day. A later phase starting in 1981, aimed to establish a Japanese capability for LWR design based on in-house technology.

The advanced pressurised water reactor (APWR) and advanced boiling water reactor (ABWR) have been developed against these utility requirements policy. Future LWRs based on evolutionary developments of these designs are being investigated by MITI and other industry groups. Mitsubishi and Westinghouse initiated in 1994, a successor programme to the APWR. Japanese BWR utilities together with Hitachi, Toshiba and General Electric (GE) initiated the ABWR evolutionary programme in 1990.

7.4.4 Korean Utility Requirements (KUR)

The Korean Standard Requirements Document (KSRD) was completed in 1990 and defines the requirements for the Korean Standard Power Plant Design, the generation of PWRs that were built in the mid-late 1990s. It has some similarities with parts of the EPRI URD but aims to reflect the wishes of the Koreans to develop their own design and construction capability.

User requirements for future plant designs began in 1993 with the objective of developing particular features and characteristics of future reactors suitable for Korea. The development of the requirements is being carried out in such a way that the requirements are being made available ahead of the design work of the Korean Next Generation Reactor.

Other requirements' documents have been produced to support specific tendering specifications, e.g. the Taiwan power company requirements document was produced to support the Lungmen project in the mid-1990s.

7.5. PERFORMANCE-RELATED IMPROVEMENTS

7.5.1 Availability

Improved performance of current plants has been discussed earlier in this book. This is being achieved by better ways of processing information on the plant condition, e.g. components, better surveillance and diagnostics. The causes of reduced level of performance can be determined by analysing the better data obtained and improved management techniques can be implemented. Clearly these types of practices equally apply to advanced as for current generation plants.

Potential improvement in performance of evolutionary plants can be established in the design phase as indicated in Table 7.1. It may also be possible to take advantage of specific improved technology, e.g. the use of high burn-up fuel to enable longer length of cycles, more advanced computer-based systems, and simpler hydrogen control systems, which require less testing during outages and thereby reduce outage time.

Other technological improvements, some of which have already been tested on current plants, concern the utilisation of better materials. For example, Inconel 690 has better corrosion resistance compared with Inconel 600 in a steam generator environment. This improved material can be used for SG replacement in current plant as well as being used for new advanced plants.

Another way, which will reduce operating costs, is to reduce the number of welds, using better forging techniques. This reduces the need for weld inspection in areas of high-radiation fluence.

Future designs should achieve improved energy availability; targets of 87% for average energy availability factor have been put forward (Juhn, 1999) for future plants. Values of high 70s% are being achieved on current plant. These figures for advanced plants can be achieved by incorporating, at the design stage, the experience gained from currently operating plant.

Table 7.1. Evolutionary plants: improved performance established in the design phase

Objective	Achieved by:
High availability: Improved design features for evolutionary plants derived from lessons learned on design limitations from current plants	Design for short outages On-line maintenance Overall simplicity of design Increased design margins
High performance: Extend performance related advances now being applied to current plants, to improve that for evolutionary plants	Improved man–machine interfaces Improved computer displays Plant standardisation Better operator qualification Simulator training

IAEA-TECDOC-1117 (1999).

7.5.2 Man–Machine Interface

Over the past few decades there has been very considerable progress in instrumentation and control (I&C) including the man–machine interface (Wahlstrom *et al*., 1999) (Table 7.2). New digital instrumentation has been developed; bringing both benefits and some difficulties. This new technology has been rapidly assimilated into conventional industry but has been incorporated to a lesser extent into the nuclear industry. The partial reason for this has been a significant downturn in the building of new plants in the last two decades of the 20th century. Other reasons are the lack of drive to replace proven old systems by new systems and in a similar vein, the conservatism of the nuclear industry and its regulators.

Nevertheless new technologies have been implemented in modernisation projects and good experience has been obtained. For new reactors, the new technology will be incorporated at the design stage. It will cover instrumentation, cabling, signal conditioning, many aspects of control, process computers and all aspects of an efficient man–machine interface. Developments relate to hardware, software, the development of information networks, interfacing and back-fitting with older systems (in the case of existing plants) and management of these aspects.

As noted above, I&C systems bring both benefits and some disadvantages. Digital systems are more flexible than analogue systems, which are limited in both practical and financial constraints. Storage capacity is not limited by physical constraints, ease of duplication of signals, better functionality of the control room, better reliability, etc. Other beneficial features are that new functions can easily be included; computers can be embedded into different components. Nevertheless digital systems are more unpredictable than analogue systems, because the software may be complex. A disadvantage of digital systems is their lack of robustness to different environmental factors such as temperature, moisture and radiation. However, commercial off-the-shelf systems can be designed to apply to the nuclear as well as the non-nuclear sector. This ensures better validation for application in some of the more challenging environments existing in nuclear plant.

Modernisation projects have been in progress in various countries – Finland, Germany, Netherlands and Sweden. Different strategies for establishing a mix between new and

Table 7.2. Evolutionary plants: instrumentation and control

Objective	Achieved by:
Utilise up-to-date technology	Transfer from analogue to digital
Overall frame of plant information management	Covering instruments, cables, signal conditioning, control room, man–machine interfaces, control equipment, process computers, real-time computers

Wahlstrom *et al*. (1999).

existing I&C systems have been developed. In Korea, for example, upgrades of the Korea Standard Nuclear Plant (KSNP) are proposed which will be implemented into the new Ulchin Units 5&6 under construction.

The I&C systems for new plant designs clearly build on the experience gained from modernisation projects on current plants. However, for new reactor designs, a more generic approach to I&C systems is being adopted. The approaches being put forward for evolutionary plant though, do not vary substantially from the more developed systems already in place on the newer present generation plants. In both cases, I&C systems are based on digital distributed systems. Control room layouts follow the approach of compactness with information displayed on visual display units (VDUs). The main future developments are likely to be simplifications in regard to redundancy and physical independence; these have been put forward in some of the more innovative designs of the future.

Differences across the reactor vendors are relatively small. The KNSR design (a typical design) implements the utility requirements of the EPRI URD, including three redundant consoles, a separated console, large display panels and additional monitoring consoles. This concept relies on the 2/4 redundancy principle. The man-made interface incorporates computerised operating procedures and the I&C design is a plant-wide digital system. The plant protection and safety control system are four-channel programmable logic controller-based systems. Non-safety controls are implemented in a two-channel system with diverse processors; similarly plant monitoring has two independent diverse systems (Wahlstrom *et al*., 1999).

7.5.3 Economic Competitiveness

Future nuclear power plant operation will have to compete with coal and gas-fired power plants, certainly for large base-load operation (Hudson *et al*., 1999). Comparative cost estimates from the last OECD study were given in Chapter 2. Within the countries that provided data, nuclear power (at the time of the survey) was found, in about half the countries, to be the cheapest option at a 5% discount rate. However, not surprisingly, at higher discount rates the nuclear option becomes less attractive.

The main factors enhancing the competitiveness of evolutionary water-cooled reactors are summarised in Table 7.3. Simplification of plant design to minimise the number of systems, valves, pumps, etc. consistent with maintaining the plant's safety envelope is a key objective. These, together with improved man–machine interfaces help to minimise operator demand and reduce risk. In the US, there has been considerable progress towards better co-operation between plant vendors and regulatory bodies in respect of the licensing process. The aim has been to develop the 'one-step licensing process'. An important objective in achieving low costs is to use a standardised approach. Thus design and engineering costs can be amortised over many units, licensing costs can be reduced, construction methods can be optimised and operator training can be made more efficient.

Table 7.3. Evolutionary plants: economic viability

Objective	Achieved by:
Reduction of capital cost:	Simplification
Lessons learned now are embodied in international utility design requirements, described in Section 7.4	Regulatory stability
	Standardisation
	Improved construction
	Multiple units
High plant availability	See Table 7.1

Hudson *et al*. (1999).

Construction duration can be kept to a minimum by adherence to the above principles. A significant fraction of the design should be completed before construction starts. The EPRI URD has introduced a quantitative criterion that 90% of design drawings must be 100% complete. Modularisation whereby plant components can be assembled in a factory helps to ensure fabrication takes place in a controlled environment, also with more automation and higher productivity.

Another way to improve competitiveness is to aim for multiple unit sites. This can be more efficient by taking advantage of better construction scheduling and the use of common administrative buildings and facilities.

Thus much can be done to improve competitiveness by reducing capital cost, which contributes to over one half of the total generation cost of a nuclear plant.

Two countries whose programmes are characterised by standardisation and technology self-reliance are France and Korea. In the case of France, large series orders have characterised the French programme. A 2% productivity gain is claimed for each unit after the second one on a given site. Similarly in Korea, for the Korean Standard Nuclear Power Plant (KSNP), the total cost of the fifth and sixth units is 15% less than that for the first and second units. For the Korean Next Generation Reactor (KNGR), a 1300-MWe PWR, there is expected to be a greater than 17% capital cost reduction compared with the KSNP.

Changes in the economic landscape associated with de-regulation of the electricity market pose particular challenges to capital intensive technologies such as nuclear energy. Flexibility in generating strategies is likely to be a requirement, e.g. building smaller size plants with relatively low investment costs and shorter pay back times. This would be coupled with a requirement for simplified technologies and infrastructure.

Concerning external costs or benefits related to electricity production costs (but not directly carried by producers or consumers), there are issues associated with job creation, resource management, sustainability and health and environmental impacts of emissions. Of these, environmental impacts are potentially the most significant. A European Commission study showed that external costs for nuclear power are lower than those for coal and gas due to the greater environmental emissions of fossil fuel plants.

For the French plants, the costs associated with health impacts were on average, 0.022 million per kWh for the current 1300 MWe plant design compared with 0.026 mill per kWh for the 900 MWe plant. For normal operation, the differences between the two types of PWR were not significant.

7.6. SAFETY THROUGH DESIGN

Already, a considerable degree of harmonisation has been achieved within the international community, on the principles of safety for commercially operating reactors. The implementation of these principles may be achieved at different levels across the countries operating nuclear plant but considerable progress has been made. Further, international safety standards will become increasingly stringent. This means that future reactor designs are likely to have to demonstrated even higher standards of safety than at present, to meet more demanding national regulatory requirements and international safety standards.

In order to do this, design principles will need to be considered for future plant (Carnino, 1999), which build on the principles already established for present generation plant. These are discussed below.

There needs to be assurance that all technical safety needs are complied with in design.

The following safety design principles are now accepted in most countries, operating nuclear plant (Table 7.4). Many of these have been put forward by the IAEA and are included in the IAEA list of 25 safety principles, listed in the next chapter.

The design must be such that plant operation is reliable, stable and manageable. Prevention of accidents is the prime goal. For many new evolutionary designs, the goal has been extended to provide better protection against severe accidents (Table 7.5).

Table 7.4. Safety fundamentals in design

Design must ensure the nuclear installation is suited for reliable, stable and easily manageable operation
Design must include appropriate defence-in-depth principle
Technology must be proven or qualified by experience or testing or both
Man–machine interface and human factors must be included in the design and in the development of operational requirements
Radiation exposures to site personnel and releases to the environment must meet ALARA principles
A comprehensive safety assessment and independent verification must confirm that the design meets the safety objectives before the operator completes his submission to the licensing authority

Carnino (1999).

Table 7.5. Evolutionary plants: safety features

Objective	Achieved by:
Increased margins and grace periods	Larger components and water volumes Lower power densities
Improved safety system reliability	Simpler redundant and diverse safety systems, greater physical separation, utilisation of high reliability components
Preclusion of high pressure core melt ejection	Reliable depressurisation systems
Increased inherent safety	Passive cooling and condensation systems
Corium confinement and cooling	Introduction of core catchers
Robust defence-in-depth	Strong containments to withstand internal and external challenges
Hydrogen management and control	Hydrogen recombiners

Juhn (1999).

The 'defence-in-depth' principle that a number of levels of protection and multiple barriers are included to prevent radioactive release is well accepted. This ensures that the combinations of failures that could occur that could lead to a significant release are of very low probability. In advanced designs, the tendency is to increase the robustness of this principle by appropriate design.

An important requirement is to ensure that the design technology is proven. Advantage should be taken of experience, if relevant, if not by further testing or possibly a combination of both.

Man–machine interfaces and human factors must be considered in the design and must be incorporated into the development of operational requirements. A key objective of newer designs is to reduce human errors.

The ALARA principle should be adopted in the design in respect of staff exposure on site and in the releases of radioactive materials to the environment. A reduction of exposures is the goal in newer designs.

Confirmation of the design via a comprehensive safety assessment and independent verification should be carried out to ensure that safety requirements are met prior to submission of the case to the regulating body.

The case must show that the risk to workers and the public is continually decreasing and demonstrate that operation is environmentally friendly.

This can be achieved by a suitable containment, which is designed to reduce the frequency of large releases to very low levels. This needs to be demonstrated via appropriate analysis (probably via deterministic and probabilistic means in addition to improved defence-in-depth).

In general, the protection of the workers and the public impacts must be demonstrated in the design, operational procedures and environmental assessments.

Development of a transparent and stable process for the licensing of plant.

A well-established and stable generating framework is an important requirement with good interfacing between the licensee and the regulatory body. The process can be enhanced via a rigorous self-assessment process coupled with independent assessment.

Need to gain public acceptance on the benefits of the proposed new design.

Harmonisation of regulatory approach, which may be more possible for new designs, is a good means of increasing public understanding and acceptance of nuclear safety.

An extremely important requirement is that there should be no serious accidents on current plants and that the nuclear industry is seen to act with integrity.

Safety requirements can be met while still maintaining costs at a level for nuclear plant to remain competitive with other generators.

The economics of nuclear power generation is improved by longer fuel cycles and by longer life (including life extension on current plants). This will clearly remain true for new designs as well.

7.7. DESIGN STATUS

Advanced plant designs are being developed to meet the requirements of utilities and regulators discussed above. They aim to provide significant improvements in performance and safety over current generation plants.

As stated earlier, advanced power plant designs are often separated into two categories, evolutionary and innovative, see for example (Juhn, 1999). Evolutionary plants are based on an evolution from an existing design through relatively small changes. The aim is to remain with design features that are proven and hence to reduce technological and other risks. Evolutionary reactors have been developed through the 1990s taking advantage of lessons learned from existing plants. These designs are, therefore, at an advanced stage of development. A number of designs have already received design certification.

Innovative designs incorporate much more radical changes in design compared with existing plants. They may include features that need verification and hence give rise to

Table 7.6. Advanced design verification to reach commercial operation

Type	Requirements	Consequences
Evolutionary	Engineering, or confirmation testing + engineering	Lower costs than innovative designs
Innovative (requiring substantial development)	Prototype and/or demonstration plant + confirmation testing + engineering Substantial R&D	Substantially increased costs

Juhn (1999).

less quantifiable risk. By definition these designs are not likely to be available for at least several decades. Table 7.6 gives some indication of the relative investment that is needed between the two categories of plant, before reaching commercial operation.

7.8. DEVELOPMENT AND INVESTMENT REQUIREMENTS

Both types of design require engineering and confirmatory testing. They may also need Research and Development (R&D) (Juhn, 1999). The amount of R&D and confirmatory testing will depend on several factors, the degree of innovation that is being introduced, the relevant work that has already been done and also the relevant experience.

These investments will be required for the design of the first plant in a line of evolutionary plants to be envisaged. For innovative plants, a prototype or demonstration plant will also be required. This latter requirement will result in the need for greater investment to meet the increasing cost of building a prototype plant as part of the development programme.

7.9. EVOLUTIONARY DESIGNS

There are a number of different evolutionary water reactor designs that are at different stages of maturity. One distinctive difference between these designs is that some incorporate established active safety systems whereas others rely on passive systems to provide some safety functions such as long-term heat removal (Yadigaroglu *et al*., 1999). More details of some of these different approaches are described in IAEA-TECDOC-1117 (1999) and Yadigaroglu *et al*. (1999).

7.9.1 Reactor Scale

New designs of plant are being proposed to cover a wide size range of power outputs. For example, large evolutionary plants with outputs of the order of 1500 MWs are being developed utilising proven active engineered systems. Medium to large-scale plants are being considered which take account of more inherently safe features such as passive safety systems. These passive systems are being scaled up for higher power output plants, in which previously the safety functions could only be accomplished by active systems. Small-scale plants are being designed which encompass novel fuel technologies, etc.

All these designs have common objectives, e.g. high availability, user friendly man–machine interface, competitive economics, and compliance with internationally recognised safety targets (Juhn, 1999).

7.9.2 Large-Scale Designs

A main issue for the future of nuclear power plant operation is capital cost of new build. One way to achieve competitive economics is to increase the unit power rating (Oka, 1999), taking advantage of economies of scale. This approach has been adopted by many of the evolutionary water reactors' designers. The power ratings of some of the proposed evolutionary LWRs may be as high as 1500–1700 MWe, many exceed 1300 MWe. Examples are given in Chapter 10. Large evolutionary designs of LWRs incorporating both proven active safety protection systems and more recently large plants putting more emphasis on passive safety systems, e.g. AP1000 are being proposed.

7.9.3 Medium/Small-Scale Designs

Through the evolution of the 1970–1980s the approach was generally to build bigger and more sophisticated reactors (Mourogov *et al.*, 1999; Anand, 1999). This approach was perceived to suffer from several disadvantages. The larger reactors were not suitable for developing countries with smaller grids. Also the increasing sophistication was not commensurate with reducing capital cost.

As a consequence, smaller and simpler designs were put forward, perhaps the best known was AP600 incorporating passive decay heat removal systems. This system has now been extended to larger scale AP1000, see above, but the approach was first introduced and verified on the lower rated AP600 design.

Another 'approach' to provide a flexible capability is to consider modular units, which can be designed, manufactured and assembled using production line processes and standardised procedures (Hatcher, 1999). The 100 MWe gas reactor pebble bed modular reactor (PBMR) is an example of this approach, introduced earlier in Chapter 2 and discussed in more detail later.

7.10. INNOVATIVE DESIGNS

A wide range of advanced reactor types has been considered over recent years but many of these would require substantial investment and development. A set of the most promising reactor types has been put forward by the Generation IV International Forum (GIF) Member Countries. Design requirements for these systems are considered later in the book.

REFERENCES

Anand, A.K. (1999) Panel Discussion, IAEA-TECDOC-1117, December 1999.

Carnino, A. (1999) Requirements and international co-operation in nuclear safety for evolutionary light water reactors, IAEA-TECDOC-1117, December 1999.

Hatcher, S.R. (1999) Global energy demand outlook, IAEA-TECDOC-1117, December 1999.

Hudson, C.R., Bertel, E., Park, K.H., Roh, T.H. & Tort, V. (1999) Economic competitiveness requirements for evolutionary water-cooled reactors, IAEA-TECDOC-1117, December 1999.

IAEA-TECDOC-968 (1997) Utility requirements and safety objectives for advanced water cooled reactors, status of advanced light water cooled reactor designs 1996, September 1997.

IAEA-TECDOC-1117 (1999) Evolutionary water cooled reactors: strategic issues, technologies and economic viability, Summary, December 1999.

Juhn, P.E. (1999) Technological readiness of evolutionary water-cooled reactors, IAEA-TECDOC-1117, December 1999.

Mourogov, V.M., Juhn, P.E. & Kagramanian, V.S. (1999) The role of nuclear power in sustainable development, IAEA-TECDOC-1117, December 1999.

Oka, Y. (1999) Panel Discussion, IAEA-TECDOC-1117, December 1999.

Wahlstrom, B., Perersen, T. & Neboyan, V. (1999) Progress in instrumentation and control including the man–machine interface, IAEA-TECDOC-1117, December 1999.

Yadigaroglu, G., Berkovich, V., Bianchi, A., Chen, B., Meseth, J., Vecchiarelli, J. & Vidard, M. (1999) Advances in technologies for decay heat removal, IAEA-TECDOC-1117, December 1999.

Chapter 8

Licensing and Safety Requirements

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Chapter 8

Licensing and Safety Requirements

8.1. INTRODUCTION/OBJECTIVES

Advanced reactors will need to meet continued demands for increased safety. This chapter reviews present legislation and possible future licensing requirements for the safety of advanced future reactor operation. The current generation of nuclear plants was designed to withstand accidents from a set of 'design basis' events. Most countries set limiting core damage frequencies and limiting probabilities for large fission product releases. An objective of many designers for advanced plants is to extend the current design basis to include accidents of increased severity and lower probability to meet expected more stringent future regulatory safety requirements.

The main focus of this chapter will be on the licensing and safety requirements for evolutionary reactors. Many regulators believe that the national frameworks already in place for existing plant remain adequate for evolutionary plant. However, there are increasing endeavours by international bodies such as the EC and IAEA to promote more harmonised agreement on nuclear safety criteria and therefore encourage a more harmonised approach to licensing in their member states. There is similar encouragement from the industry side with the development of standardised utility requirements (URs) for member states, e.g. the US and European URs described in the previous chapter.

8.2. INTERNATIONAL SAFETY PRINCIPLES

Laws and statutes exist in most countries to ensure the safe operation of nuclear plant, see, e.g. EUR 20055 EN (2001) and EUR 16801 EN, ISSN 1018-5 (1996). Health and Safety laws are defined by government ministries, taking advice from various other supporting organisations. Safety standards are enforced by Safety Authorities and Regulators who grant licences for operation in accordance with national laws. These are reinforced by various international bodies, e.g. IAEA.

Internationally, IAEA principles have been established that govern the relationship between the regulator and operator. These are summarised in Table 8.1. These principles are embodied in the regulatory requirements of most countries.

In particular, these principles have played considerable influence in furthering the progress in the EU Enlargement countries from a closed safety culture to one of greater openness. Progress towards generally accepted international standards has also been

Table 8.1. IAEA safety principles (abbreviated form)

1.	National governments shall establish a legislative and statutory framework for regulation
2.	Prime responsibility for safety is assigned to the operator
3.	Independence of the regulatory body from the operator
4.	In all activities, safety matters have the highest priority
5.	Establishment and implementation of appropriate Quality Assurance (QA) programmes
6.	There are sufficient available adequately trained and authorised staff
7.	The capabilities and limitations of human performance must be recognised
8.	Emergency plans for accident situations must be in place and appropriately exercised
9.	Site selection must take account of all relevant features affecting safety
10.	The design must be suited to reliable, stable and manageable operation
11.	Design shall include appropriate application of the defence-in-depth principle
12.	Design technologies shall be proved by experience or testing or both
13.	Man-machine interface and human factors shall be considered in design and operation
14.	Radiation exposures to site personnel and to the environment shall be ALARA
15.	The design shall be confirmed via comprehensive safety assessment and independent verification
16.	Specific approval of the regulator is required prior to the start of operation
17.	Operational limits must be defined from safety analysis, tests and subsequent operational experience
18.	Operation, inspection, testing and maintenance must be conducted by adequately trained and authorised personnel
19.	Competent engineering and technical support to be available throughout installation life
20.	Documented procedures must be established for anticipated operational occurrences and accidents
21.	All plant operational incidents significant to safety must be reported to the regulator
22.	All radioactive waste must be kept to a minimum (both in terms of activity and volume)
23.	The design and decommissioning programme shall aim to limit exposures during decommissioning to ALARA
24.	The operator shall verify by analysis, testing and inspection that the physical state of the instillation remains in accordance with operational limits
25.	Systematic safety assessments shall be performed throughout life

Govaerts (1996).

influenced by other international bodies, e.g. OECD and the EC (within the EU and EU Enlargement countries).

8.3. SAFETY INFRASTRUCTURES

Governmental and regulatory infrastructures have to a large extent developed in parallel with national nuclear power programmes, certainly in Western Europe, US and Japan. The independence of regulatory bodies from the organs of government or private industry promoting nuclear power is an important requisite that is now generally internationally accepted. Countries in former Eastern Europe have made significant progress in establish their own independent regulatory bodies, during the 1990s, having previously relied on the centralised systems of the former Soviet Union.

IAEA Basic Safety Standards were established in the mid-1990s to ensure the safety of all applications of nuclear technology, particularly industrial and medical applications. In some countries, these had developed without adequate infrastructures to ensure the safety of these applications (IAEA/NSR/2002, 2003).

As stated in the IAEA principles earlier, one of the tenets for a strong independent regulator is the availability of an adequate pool of qualified staff. As noted in Chapter 2 with declining nuclear programmes in some countries, there are fewer qualified engineers available to regulatory bodies who frequently seek engineers who have acquired on-site experience in industry.

To meet these requirements, the IAEA has instigated various education and training programmes. These aim to promote self-sustaining capabilities in the member states, at all levels, national and regional. These include programmes to train trainers, disseminate materials and harmonise on-the-job training programmes. They are also establishing centres for education and training, centre networks and exploiting modern technology for distance learning and e-learning.

There is a large amount of information available on the safety and operation of nuclear power plants (NPPs), which has not been fully disseminated worldwide. Networks are being developed to share this information and provide a means of mutual sharing of information. International bodies including (e.g. IAEA, EC, CSNI) act as facilitators in various ways with regard to sharing this information.

8.4. NATIONAL REGULATORY FRAMEWORKS

The status of selective national regulatory frameworks in relation to the design and safety of future NPPs is reviewed below (IAEA-TECDOC-905, 1996).

8.4.1 UK

In the UK for example (EUR 20055 EN, 2001), safety is governed by the Nuclear Installations Acts 1965 and 1969 (NII Acts) and by the 'Health and Safety at Work Act 1974 (HSW Act)'. These are supplemented by the Nuclear Installation Regulations, the Ionising Radiation Regulations and other Licensing Conditions. Regulatory Guides (non-mandatory) include the Tolerability of Risk (TOR) for Nuclear Power Stations (HSE, 1992) and the Safety Assessment Principles (SAPs) (Harbison, 1992). These latter two documents are to provide guidance to NII Assessors in assessing Licensees' safety cases but are not legislative.

The Health and Safety Commission (HSC) is responsible for preparing proposals for safety laws and standards approved by the Secretary of State for Environment (DOE). It advises the Department of Trade and Industry (DTI) regarding regulatory matters in England and Wales and the Secretary of State for Scotland. The HSC is advised by the

Advisory Committee on Nuclear Installations (ACSNI) and the National Radiological Protection Board (NRPB). The HSW Act is enforced by the independent UK Government Health and Safety Executive (HSE), under the HSC. The HSE is responsible for granting nuclear licences and the enforcement of the Health and Safety Laws. Licences and Inspections are administered by the UK NII who have the authority to withhold licences for nuclear plant operation.

8.4.2 US

Additional to its current nuclear generation commitments, the US has in place programmes for the design and potential licensing of advanced light water reactors (ALWRs). Various ALWR designs have been certified in readiness for construction in the event of a new build programme.

The important codes, guidelines and URs are set down in a prescriptive set of documents that encompass existing reactor regulations, 10 Code of Federal Regulations (CFR) 50 (USNRC, 10 CFR Part 50, 1988). A new regulation, entitled 'Early Site Permits; Standard Design Certifications; and Combined Licences for Nuclear Power Reactors' was published in 1989 (USNRC, 10 CFR Part 52, 1989). This has been used in the process of issuing design certifications for the ALWR designs.

In addition to these codes, various USNRC policy statements have been issued on standardisation, regulation of advanced NPPs, goals for safety and severe accidents. The USNRC have also issued safety evaluation review reports, NUREG-1242 (USNRC Review of Electric Power Research Institutes, 1992–1994) on the EPRI ALWR URDs (EPRI NP-6780, 1990), discussed in Chapter 7.

The USNRC have established a number of important principles regarding the safety of future reactors. In general, future reactors should achieve a higher degree of safety than is deemed acceptable for currently operating plant. Severe accidents need to be considered in the design process. The US requires a complete Probability Safety Assessment of the design (as do the URs Documents that are more conservative than the USNRC goals for current generation plants by at least a factor of 10). Further accident management measures need to be identified during the design process and should be an important mitigation in severe accidents.

8.4.3 Finland

The Nuclear Energy Act and a supporting Nuclear Energy Decree 1988 cover the construction and operation of nuclear facilities and all other matters in connection with the management and handling of nuclear materials and nuclear wastes in Finland (EUR 20055 EN, 2001). Additionally there is also the Radiation Act and Decree 1991 that is applied to the use of nuclear energy. Various Ministries have the responsibility for nuclear energy safety and security. The Radiation and Nuclear Safety Authority (STUK) is the primary regulatory body. It is an independent governmental organisation for the regulatory control of radiation and nuclear safety. There are also several acts (Act 1069/83 and Decree 698/97) that enforce the responsibilities of STUK.

The regulator does not specify particular design codes but there are guides that set down the requirements for the design of NPPs. The regulatory system is based on a comprehensive system of regulations and safety guides but it allows for the further development of safety culture within the industry. The current system is considered adequate for the licensing of future evolutionary LWRs.

8.4.4 France

The fundamental legislation for nuclear energy in France is based on the Decree on Nuclear Installations issued in December 1963, together with further decrees in 1970, 1974 and 1984 (EUR 20055 EN, 2001). The regulatory body is formed within the Ministry for Industry and administered by the Ministry for Environment. It is represented by DSIN (Direction de la Sûreté des Installations Nucléaires) which is responsible for regulation and inspection of the plants.

The regulatory regime is not prescriptive; no particular design codes are prescribed by DSIN. However, Basic Guidelines for Safety RFS (Règles Fondamentales de Sûreté) are defined by DSIN. In practice, American design codes were used (ASME) but later, French design codes (RRC: Règles de Conception et de Construction) have been developed by the French industry that meet the requirements of the safety authority.

In 1989, France and Germany agreed to harmonise their safety approach for future reactors. In 1990, the safety authorities of both countries formed the DFD (Deutsche-Französische Direktion) forming close links between DSIN on the French side and BMU on the German side (Bundesministerium für Umwelt, Naturschutz und Reaktorsicherheit). In 1992, the DFD agreed that the DSIN and BMU would establish a common safety approach for future reactors.

8.4.5 Hungary

The main legislation for nuclear safety in Hungary is the Act on Atomic Energy (Act No. CXVI of 1996 on nuclear energy), which became law in 1997 (EUR 20055 EN, 2001). For implementation of the act, there are a number of regulations, 12 Government Decrees and 33 Ministerial Decrees. These are issued by various ministries: the Hungarian Ministries of Interior, Health, Agriculture, Economic Affairs, Transport and Water Management and Environment. The primary regulatory body is the Hungarian Atomic Energy Authority (HAEA).

The Hungarian safety regulations are generally non-prescriptive. There are no particular design codes defined by the Authority. However, certain requirements are set, e.g. in regard to the validation of methodologies used, etc.

The nuclear safety regulations have been set fairly recently over the past decade. At the present time, they are considered to be appropriate for future reactors as well.

8.4.6 Russia

Nuclear safety in Russia is governed by laws on radiation safety of the population of the Russian federation, a law on radioactive waste management and a law on the utilisation of atomic energy (IAEA-TECDOC-905, 1996).

The regulatory standards that are applied to nuclear plants in Russia are described in a basic document 'Atomic Power Plants General Safety Regulations', OPB-88 (IAEA-TECDOC-905, 1996). This is supplemented by a number of documents on basic regulations for the assurance of safety in nuclear plants (OPB-88), PNAEG-1-011-89; radiation safety standards, NRB-76/87; nuclear safety rules for NPPs, PBYa-Ryu-AS-89, PNAE G-1-024-90; and rules of design and safe operation of equipment and piping of NPPs, PNAE G-7-008-89. In 1993, a regulation was issued by the Federal Nuclear and Radiation Safety Authority of Russia on NPP Siting to deal with limiting the consequences of severe accidents and requiring that they be considered in the design of future reactors (Federal Nuclear and Radiation Safety Authority of Russia, 1993).

There is a large work programme in improving the safety of currently operating plant to ensure their safety standards are consistent with latest regulations.

8.4.7 Japan

Current plants in Japan are licensed against Japanese regulatory standards, codes and guidelines. The Japanese approach to safety is harmonised with other international activities through participation in IAEA activities. For example, Japanese utilities participate in the US ALWR initiatives.

The requirements for future ALWRs are being established by both regulator and utilities. In parallel, design programmes for future plants are proceeding with improved safety levels. Current Japanese plants such as the APWR and ABWR have core melt frequencies below 1E-6/reactor year, below the International Nuclear Safety Advisory Group (INSAG)-3 target for future plants of 10E-5/reactor year.

8.4.8 General

In regard to different national practices, authority may be administered at either a national or local (federal) level depending on the laws of the country concerned. There are differences in approach reflecting differences in local legislation (EUR 16801 EN, ISSN 1018-5, 1996; Table 8.2). In the UK, the licensing process for power plant lies with a single licensing authority, the NII. However, in Germany for example, regulatory responsibilities lie with the Ministries of the Federal States. In addition, the licensing of different sectors of the nuclear industry may be administered across several organisations.

There are significant differences across countries on how regulatory technical support is acquired. Few authorities have sufficient in-house technical capability to meet all their requirements within their own organisation. Certain authorities require work to be contracted out to other organisations. In some countries, technical support organisations

Table 8.2. Differences in regulatory frameworks and approach

Framework/activity	Approach
Administration of authority	National or local
Regulatory regime	Prescriptive or otherwise
Technical support procurement	Maintenance of government-funded TSOs vs. services purchased from commercial companies
Licensing and safety culture	Open or closed
Utility/regulator relationship	Collaborative vs. formal approach
Conduct of research	Focus on current operational or more future needs

EUR 20055 EN (2001).

(TSOs) are supported within the countries' national safety framework. This is the case, e.g. in Germany and France. In other countries, the regulator may call on various contractors; (usually privately owned or commercial companies) to supply his technical need for the particular work required. This has implications on how an adequate level of technical resource is maintained and indeed on the number of personnel available. There are also substantial differences in the levels of resource (both licensing and technical) available within regulatory authorities.

In the past, the approach to nuclear regulation has varied between national governments and their political persuasions. For example, in the countries operating Russian-designed plant under the former Soviet regime, the safety culture was very different from that of the West. The differences of approach are now much less marked as all these countries move towards common licensing approaches.

There has been a major global influence of US practices in many countries. This is because US-designed plants are in operation in many countries around the world. It has also been a common principle from many regulators that NPPs should be licensable in their country of origin.

In addition, there has been considerable influence from IAEA principles, (Govaerts, 1996; IAEA Safety Standard Series, 2000), which have promoted the safety culture and co-operation between regulator and operator. These cover commonly accepted principles such as operator responsibility for operation and quality assurance principles agreed between regulator and operator, etc.

As a general rule, the investment in nuclear safety has far exceeded that which has been devoted to other industrial operations. As a consequence, the safety standards in the nuclear industry are as high or higher than those existing in many other industries.

Clearly, the licensing of future plants may require some adjustments of regulatory approach. For example, an application for the licensing of a Generation IV system would introduce new safety issues associated with new technology. The further the new system had advanced from existing technology, the greater the regulatory adjustment that would be required.

8.5. DESIGN BASIS

Many of the principles for design basis assessment have been established for present-day reactors over many years. These include the 'defence-in-depth' principle, needs for diversity and redundancy, Safety Analysis Report (SAR) assessments and so on. In this section, the principal design basis approaches that are likely to apply to reactor licensing in the future are reviewed.

Most of the licensing submittals for present-day plant have been submitted using the conservative evaluation model (EM) methodology (Table 8.3). Conservative modelling was required to overcome lack of detailed knowledge of the phenomena. This methodology is commonly referred to as 'Appendix K' referring to the relevant appendix in the US CFRs (10 CFR 50).

BE methods are likely to be a common goal for licensing in the future, including those for future reactors. BE methods have been accepted by the USNRC (and other regulators) c.f. Appendix K Revision in 1988.

The acceptance criteria for fault studies are established by good understanding of the physics of present designs. However, as designs evolve, these criteria may need to be re-evaluated. The additional changes may also be required to accommodate extensions in, e.g. mode of operation.

There are likely to be increased requirements for Probabilistic Safety Assessment (PSA) studies (Table 8.4) to underpin deterministic studies and to help estimate doses to the population, source term of release, etc. Probabilistic targets are likely to become more

Table 8.3. Licensing methodologies

Methodology	Description
Evaluation model (EM)	Conservative modelling
Best estimate (BE)	Physical models without bias
Risk informed (RI)	Approach depends on relative risk but this concept is only at the development stage

EUR 20055 EN, (2001).

Table 8.4. Safety approach for severe accidents

PSA approach	Selection of most probable sequences leading to a core melt Provision of preventative or mitigative measures Wide coverage of possible sequences Good quantification of the benefits from proposed measures Depends on the status of PSA accident analysis
Deterministic safety analysis approach	Definition of containment challenges from core melt behaviour Assurance of containment integrity by design measures

Wahlstrom (2003) and Crech (1999).

stringent, e.g. on core damage frequency or on containment limits. The current trend is to use best estimate methods for the frequencies and probabilities in PSAs.

The move towards BE methods is being supported by regulators and utilities because more realistic margin estimates enable a better quantification of actual risk to be obtained and enable a wider operating window.

However, there are developments required before the methodology is likely to be regarded as a mature engineering tool. It is necessary to be able to quantify unbiased uncertainty limits on key parameters (e.g. peak clad temperatures) and as yet the methodologies are not yet very practicable for licensing studies.

Another factor is that there is generally reluctance to change from an established methodology that is accepted by all parties.

Looking further to the future, risk informed methods (Wahlstrom, 2003) are being put forward by the USNRC but these methods require further development. Traditionally, the safety of NPPs has been justified by a deterministic approach based on the defence-in-depth principle and single failure criterion for design basis accidents, etc. Probabilistic approaches have further developed and now the probabilistic safety analysis methodology is becoming well established. These provide a means of taking a systematic approach to determining the probability and therefore risk of various failure sequences.

The idea of risk-based or risk-informed approaches is to focus on the most important issues in terms of risk. If PSAs are used to determine the risk then clearly it is important that there is confidence in the PSA methodology used. Risk-informed approaches can be applied to new reactor design or indeed to assist modifications of old plants, to target maintenance actions and inspections. PSA methodology can also be used to identify the safety categorisation of components.

The USNRC has made a commitment to move towards a risk-informed regulatory regime. Other regulators are considering the development of the approach.

The notions of design basis and defence-in-depth have been well established in the licensing of present generation reactors. For some future systems, these notions may need to be revised in the light of newer technologies with very different designs, materials and fuel cycles.

8.6. SEVERE ACCIDENT APPROACHES

In reactor safety research, there has been a continued drive to improve the understanding of severe accidents in order to prevent significant releases of radioactivity under severe accident conditions. The work has been focussed on different levels corresponding to the defence-in-depth principle discussed earlier.

The first objective during a core melt accident is to maintain vessel integrity following attack by molten corium or debris from the higher up reactor core internals. The corium may or may not be in contact with water and even in the latter case may not coolable. Various research programmes have been carried out including investigation of early and late phase melting phenomena under different accident conditions and the energetics of corium/water interaction studies of heat transfer-related mechanisms from the debris to the vessel. Others include thermal–hydraulic cooling of debris beds, investigations of the structural response of the vessel and examination of the effectiveness of cooling of the vessel with water from the outside, etc. Theoretical programmes of work supported by experimental programmes have been carried out. The combined programmes have considered scaling effects and also how to extrapolate results obtained from simulant materials to reactor materials.

If the vessel is breached, then molten debris will be released in the cavity beneath the reactor. It is therefore important to understand the physical processes of the potential release of melt from the vessel, how it will spread over the concrete floor or how it might be impeded by other retention structures (in some of the newer designs). It is likely that there will also be interaction of corium with water from discharge of emergency core cooling systems (ECCS) and the need to quantify the load on the containment from any resulting steam explosions.

The EC programmes have covered experiments and theoretical studies on the thermochemistry of molten corium interactions with structures. They have included projects to determine the production of hydrogen and other non-condensable gases, e.g. carbon monoxide to establish the threat to containment from gas combustion. Another facet was to consider the retention of fission products in the melt with respect to the 'Source Term' (see below). Work items covered vessel failure and corium release modes, corium spreading effects and the consequent impact on direct containment heating. It also covered the interactions of corium with water and structures and generic studies on retention devices (e.g. core catchers).

An important aspect of severe accident research has been to quantify the Source Term. This is defined to be the quantity, timing and physical form of the radiological and chemical species release to the environment. It is dependent on the type of accident. The important inputs for determining the Source Term are fission product release from the fuel, and the transport in the primary circuit and the containment. Also important are the suspension, resuspension and condensation/revaporisation mechanisms within the reactor circuit and containment. Accident mitigation devices such as sprays and other measures have an important mitigation effect (Table 8.5). There are important large-scale integral tests and supporting separate effects tests to provide data to validate computer codes for analysis.

The ultimate Source Term to the environment depends on whether the containment is breached. There are different threats in the short term and long term.

Assuming the containment holds the Source Term will depend on the leak tightness of the containment. The short-term threat arises from corium/steam explosion, hydrogen

Table 8.5. Example of severe accident sequences and mitigation

Sequence initiator	Consequential failures	Consequence mitigation
SBLOCA	Failure of HHSI, failure of rapid secondary depressurisation	Depressurisation. PARs and use of sprays
SLB + SGTR	Late failure of both trains of SI and sprays at recirculation	Refilling of the CST Primary depressurisation via PZR valves
V LOCA into auxiliary building	None	RCS depressurisation, hydrogen recombination, pipeline retention
Reactor trip with unavailable MFW	Loss of all FWs, AFW and EFWs	Hydrogen recombiners
Transient	High-pressure ECCS, ADS	Containment and vessel venting Containment flooding
Transient	Total loss of off-site power	Manual filtered venting

Ang *et al.* (2001).

combustion, more particularly detonation, direct containment heating, secondary effects of missiles and containment isolation failure. The long-term threat comes from the build up of heat (and therefore pressure) due to failure of removal of decay heat or by failure of isolation devices, i.e. material failure. The containment strength is very design dependent and different mitigation systems will be feasible for different designs.

Within the EC framework programme, there have been generic experiments and theoretical studies on hydrogen combustion (deflagration and detonation), thermal-hydraulics, stratification and natural convection. Also included in the programme has been investigation of dynamic concrete behaviour at high impact velocity and studies of leakage of steam and aerosols through cracks and penetrations. These have been in conjunction with the identification of mitigation measures.

8.7. SAFETY STANDARDS

There may be differences in the regulations and standards for future reactor licensing. In countries favourable to nuclear power, the licensing frameworks are likely to evolve or be extensions of existing frameworks for currently operating plant. However, there are a number of countries that are not favourable towards nuclear power where the approach will be dependent on the perception of relative risks to benefits. There are also a number of countries where no new plants are planned or where moratoria are already in place in which case there is no issue. Within Europe for example, five out of the eight EU member

Table 8.6. Standards for future reactors (in countries where there are not moratoria)

Current regulations and standards are likely to be appropriate for evolutionary type plant
Current regulatory regimes may need to be extended for more revolutionary type plant
Future reactors are expected to include greater protection against severe accidents. Increased level of PSA
Increased use of BE methods to demonstrate more realistic safety margins
Increased use of Risk Informed methods following USNRC lead

EUR 20055 EN (2001).

states with nuclear power are in this category. These include Belgium, Germany, Netherlands, Spain and Sweden.

The standards for future reactors will depend on internationally accepted standards, e.g. as shown in Table 8.6. Some countries believe that their current regulations and standards are already appropriate for future evolutionary LWRs. There are, however, certain requirements that are likely to be imposed which assume greater significance for future reactors.

It seems likely that future reactors will have to include greater provision against severe accidents. This may be required to be demonstrated by both PSA and deterministic means.

PSA methodology is used currently on existing plant to identify weaknesses and therefore enable modifications to be implemented. PSA practices have improved significantly over recent years; these methods provide an accepted means of assessing the safety of a plant. PSAs provide a means of verifying the design basis of a plant supported by deterministic analysis making conservative assumptions. They also can be used to assess whether there are any 'cliff-edge' concerns about safety with the design, e.g. just beyond the design basis. This has led to modern LWR designs, which restrict the source term for radioactive release for beyond design basis, including severe accidents.

There is certainly a trend in some countries to extend the design basis for new plants to cover severe accident challenges. However, this could clearly have a major impact on competitiveness of the plant, through the cost of including specific or additional components.

From a severe accident perspective in LWRs, the strength of the containment is crucial in limiting radioactive release. However, at present, the containment is only built to withstand DBAs where safety systems are assured, and assumed to respond subject to a single failure criterion. Present analyses demonstrate a margin between the design pressure and the actual failure pressure and this is useful in evaluating the implication of certain severe accidents. However, if all severe accidents were included within the design basis envelope, then the containment would need to be strengthened to withstand higher loads.

There are differences worldwide in the approach to containment and its function in severe accidents. For example, within the IAEA member states, some countries have already made significant improvements, others have plans for improvement that have not yet been implemented, others prefer to adopt a different approach to severe accident management.

The extension of the design to include severe accidents has been proposed in Germany. However, at the time of writing, there is a consensus to terminate the use of nuclear energy over the next 30 years and therefore standards for future plants are no longer under consideration. Other countries, e.g. France are taking a similar position. If such proposals are adopted, there could clearly be major differences in approach to licensing across the nuclear operating countries.

In many cases, the evolutionary designs contain more advanced safety features, some of which already mitigate against severe accident vulnerabilities. The EPR design, for example, has a debris retention component. Some VVER-1000 reactors' future designs will adopt a similar approach. Thus, the addition of core catchers to prevent melt attack of the containment base-mat is one feature that has already been introduced into the design to cover severe accidents.

It would, however, be very difficult to extend the design basis to cover all potential severe accident scenarios. Steam explosion vulnerabilities are still uncertain and it would be difficult to demonstrate by deterministic means that a containment is sufficiently strong to withstand all possible loadings, taking account of the uncertainties.

Other approaches on design have been to take advantage of more inherent mechanisms, passive injection, gravity driven flow, e.g. as in AP-600.

A chapter is devoted to passive plants later in the book.

8.8. FUTURE REACTOR DESIGN STRATEGIES

Future plants may be of a number of designs since there is no general agreement on what features should be included in future designs. Further, there is no universal agreement on the design basis and how improvements in safety can be quantified. Further there are worldwide differences in licensing positions and engineering design standards across the world.

There has been progress in Germany and France towards developing harmonised approaches to design and licensing, e.g. within the EPR initiative.

In the US, the design of the AP600 has been certified by the USNRC as meeting accepted standards. This provides a demonstration to a potential regulator that the design has been certified against a particular standard. Generic design requirements can be derived from IAEA standards. These provide a norm for vendors to demonstrate how their particular designs meet these requirements.

Future designs will also have to satisfy URs and the evidence to date is that there are different vendors proposing a wide range of designs in the market. Different regulating bodies may be sympathetic to different designs and indeed different safety solutions for the same design. A harmonisation of design requirements and safety solutions (if they can be agreed by regulators) would clearly be desirable for vendors who could then seek design certification that would be acceptable in a number of different countries. For the same reasons, it would be an advantage for utilities.

Since there are likely to be operator applications for quite different designs in the future, there would clearly be a benefit in a more ‘technical neutral’ approach to licensing if it could be acceptable to the regulator, i.e. the licensing process would become less design specific than it is today.

8.9. HARMONISATION OF REGULATION

8.9.1 Existing Plants

Many LWRs operating in the Western world were designed according to US safety criteria and philosophy based on the defence-in-depth principle in design. This also includes the construction, maintenance and inspection and operational practices that were developed according to the US model. Some countries introduced country-specific regulations, e.g. associated with higher density populations or the requirement to withstand military aircraft crash, i.e. as in Germany. Nevertheless, the US historical influence has tended to encourage a process of harmonisation in the regulation of LWRs.

Within the Western world, there has always existed openness in communication at the level of research. This has had the result that significant differences in practices have been discovered. Regulators have been informed and the most advantageous common approaches adopted in many cases.

There are some areas where there has been a smaller level of harmonisation due to differences in safety philosophy because of redundancy requirements, levels of conservatism, etc. It is also difficult to develop a harmonised approach to safety criteria because of differences in plant design. Two particular areas are in the fields of fuel safety criteria and PSA.

In general, there is a greater degree of harmonisation in operational safety, in the requirements for Non-destructive Testing (NDT), on environmental qualification, the benefits of periodic safety reviews and the merits of risk based service inspection.

Some of the benefits of harmonisation for future plant are given below and also in Table 8.7.

8.9.2 Future Plants

There should be greater scope for a harmonised approach to licensing new designs. Already, this is happening for evolutionary reactors. Many of these have been designed against URs.

Table 8.7. Potential benefits of a harmonised approach to licensing

Achievement of a common licensing position across a number of countries would increase the common market
Common international standards for plant design would facilitate the licensing of plant
Harmonisation of design requirements, enabling design certification would benefit vendors
A harmonised regulatory approach would benefit utilities by reducing uncertainties in the licensing process

EUR 20055 EN (2001).

For example, designs have been specified by utility companies in Europe in consultation with regulatory authorities. A good example of this approach is the EPR French–German co-operation. Another co-operation involved the Westinghouse 1000 MWe passive plant reactor development programme, the Siemens 1000 MWe BWR and the Westinghouse Atom BWR90+.

Harmonisation of EURs provides a focus for utilities and is more important than harmonisation in international working groups, which may be less focussed on utilities' specific requirements.

Harmonisation can result from a design certification process, as adopted for AP600 and which is in progress for AP1000.

Another means to improve harmonisation would be in the common development of standards but this is not currently the situation.

Harmonisation of approach has resulted in increased consideration of severe accidents at the design stage.

8.10. INTERNATIONAL COLLABORATION

8.10.1 IAEA

There is a broad ranging programme within the IAEA (EUR 20055 EN, 2001; IAEA, 2001; IAEA, 2000a; IAEA, 2000b), to promote safety in civil nuclear power reactors within its member states. The programme covers safety standards for nuclear reactors, radiation, waste and transport safety. It includes publications of Codes of Practice that establish the objectives and minimum requirements for the overall safety of NPPs. They cover topics such as: the regulation of NPPs, safety in the siting of plants, design for plant safety, safety in plant operation and quality assurance for safety.

The IAEA also publishes specific Safety Guides within its National Safety Standards (NUSS) programme. These Guides and the above Codes of Practice are recommendations issued by the IAEA for use in its member states but there is no requirement in general for a country to adopt these NUSS standards in legislation. However, the IAEA's safety standards are endorsed by its member states, including the EU member and candidate states, as representing best international safety practice. Many documents have been published on safety fundamentals, requirements and guides. The NUSS programme covers the following areas: general safety including emergency preparedness and response and legal and governmental infrastructure, safety in the design and operation of plants, radiation safety, radioactive waste including discharge and disposal, and transport safety.

Significant recent safety standards publications in 2002 include safety requirements on preparedness and response to a nuclear or radiological emergency (IAEA/NSR/2002, 2003; IAEA Safety Standards Series No. GS-R-2, 2002). Additionally new and revised safety guides have been published in 2002 on legal and governmental infrastructure, a number on various aspects on power plant safety and one on radiological protection of patients and on the management of mining and milling waste.

The safety of the transport of radioactive material in a continuing priority for IAEA. The IAEA has introduced its transport safety appraisal service (TranSAS) to ensure that the agency's transport regulations are consistently implemented across the member states and is fostering a greater degree of transparency and collaboration. With the increased security concerns following 11th September 2001, there is a reduction in commercial carriers that are available to carry radioactive sources or material. An international conference on the safety of transport of radioactive material is scheduled for July 2003 (IAEA/NSR/2002, 2003; International Conference on the Safety of Transport of Radioactive Material, 2003).

An important part of IAEA strategy is to co-operate with other international bodies such as the OECD/NEA and WANO, in addition to facilitating technical co-operation with developing countries.

The IAEA has a number of programmes directed towards future nuclear power development and applications (IAEA Technology, 2002). The potential of new innovative reactors has been reviewed in a co-operation with IEA and NEA of the OECD. The conclusions of the study have been provided to the US GIF, see below. Regarding other recent programmes, the IAEA initiated its International Project on Innovative Nuclear Reactors and Fuel Cycles (INPRO) in 2000. There are other initiatives, e.g. on passive cooling in evolutionary LWRs, super critical water reactors, HTRs, technologies for waste incineration, nuclear heating applications and desalination.

8.10.2 OECD/NEA

The OECD/NEA helps its member countries in the maintenance and development of the legal, economic, scientific and technological frameworks needed for the safe and economical use of nuclear energy (OECD/NEA, 1999; OECD/NEA, 2000). It was established as early as 1958 and the current membership extends to 27 member countries. It aims to facilitate a harmonised approach to resolving safety issues. It aims to include the experience of the Central European countries that continue to operate Russian-designed plant and also Russia itself.

The NEA is based on a Technical Committee Structure, led by a Steering Committee. The latter provides guidance on the direction of work that is the executed by the former, often with the assistance of specialist Working Groups and/or Expert Groups. A number of committees have been formed over the years covering nuclear regulatory activities, safety of nuclear installations, radiation protection and public health, radioactive waste management, nuclear development and fuel, and nuclear law.

These Committees and their Working Groups provide access to information, enable common experience to be passed around, promote the convergence of technical issues and generate help to promote areas of common interest. They also encourage co-operation with the IAEA including the member states of the EU, US, Japan and Russia, which is part of the OECD/NEA strategic plan.

The OECD/NEA are facilitating initiatives in innovative reactor study programmes in collaboration with IAEA, as noted above. A whole range of current programmes is listed in IEA/NEA/IAEA (2002). These include wide-ranging activities on small–large scale designs, the range of different systems in GIF, these being considered from all aspects including safety, performance and economics. Along with IAEA, there is collaboration within the Michel Angelo Network (MICANET) of the EC.

8.10.3 EC

EC activities have been performed over the last few decades to meet the requirements of several resolutions (EUR 20055 EN, 2001). These were the Council Resolutions of 1975 (European Commission, 1975) and 1992 (European Commission, 1992).

The 1975 Resolution stated that European Community actions in the area of nuclear safety were necessary because of the importance of nuclear power as an energy source in the Community, the need for the Community to address the technological problems of nuclear safety in view of possible environmental and health implications, the need to keep the public informed, to realise the safety and economic benefits of a harmonised approach for nuclear safety authorities, constructors and producers, and the desire for the Commission to influence global nuclear safety.

The 1992 Resolution not only acknowledged the continuing importance and relevance of the earlier resolution but also recognised some additional requirements. For example, the Council reaffirmed the importance of progress, nuclear safety research and innovation including future generations of reactors, but recommended that experience gained should be extended to third countries, particularly those of Central and Eastern Europe and the republics of the former Soviet Union.

Since 1995, there have been further developments, including the 1995 Consensus Document on European LWR safety, the publication of recommended licensing procedures, a document on the implementation of the 25 Principles of Nuclear Safety in different EU countries and the Convention on Nuclear Safety by EU member states and the EC. The EC is encouraging the spirit of common approaches to nuclear safety via dialogue and the synthesis of information. Nevertheless, at the present time, there are no Euratom

Treaty obligations on the EU member states to harmonise their nuclear safety criteria and regulations.

The EC Euratom framework programme for future reactors is linked with the national and international programmes above (Ion *et al.*, 2003). Additionally, there is a major investment in fusion. With regard to fission reactors, the EC MICANET objective is to provide an R & D strategy to enable the nuclear option to be kept open via the development of innovative systems. Euratom also participates within the GIF project.

8.10.4 Generation IV International Forum (GIF)

A group of 10 countries (Argentia, Brazil, Canada, France, Japan, Republic of Korea, South Africa, Switzerland, UK and US) are working together specifically to develop a roadmap to pursue R & D on future Generation IV systems (Generation IV Nuclear Energy Systems, 2003). These innovative reactor systems are described in subsequent chapters. The GIF initiative is relatively recent, starting in January 2000 and initiated by the United States Department of Energy (USDOE). The objectives are to develop future reactor systems that are competitively priced, while addressing safety, waste, proliferation and public perception concerns. The reactors cover water, gas-cooled thermal and fast spectra, liquid metal (sodium, lead and lead–bismuth) cooled and molten salt designs.

8.11. EVOLUTION OF SAFETY

This section considers the question on how safety standards are likely to evolve in the future. This is difficult to forecast in the present climate where in many countries in the Western world, there are few active initiatives for new build. Moreover, there are a large number of plants still currently operating of very different ages and very different design types. There has been a significant investment in upgrading certain classes of reactors, e.g. some of the older Russian-designed plants in Central and Eastern Europe, the early VVERs and RBMKs have a much more limited design basis than would be acceptable for licensing today.

There is general acceptance, at least within the nuclear community that most operating reactors in the world are safe. This is not the view of a large portion of the general public and the tendency will be a drive to continue to make reactors safer.

There is international pressure to improve and unify safety standards across the world. The IAEA standards are generally recognised as the starting point for most countries. These are generally taken as the basis for national standards in most countries, particularly in those with less well-developed nuclear regulatory frameworks.

There are also EC initiatives to develop common approaches in safety across Europe and the EU accession countries. There has resulted in a more unified approach across Europe with regard to regulator approaches; e.g. one principle is independence of the regulator from the licensee. However, there are limits in the level of unification that can be achieved at the technical level across such a diverse range of types of reactor in operation. Some plant component upgrades are possible to bring plants up to a higher level of safety commensurate with more modern plants, but it is difficult to improve the safety level of some elements of the design, e.g. strengthening of the containment.

The majority of proposals for new reactors has been ‘evolutionary’ water-reactor designs. These have a similar level of safety to that found in the more recent current generation plants but the evolutionary designs have some additional safety features. These include the use of passive systems as in the AP600 and 1000 systems, also the availability of ex-vessel flooding capability in these plants and core catchers as in the EPR and the latest VVER1000.

However, for other reactor types the situation is much less clear. The requirements for the design basis of LMFBRs are much less clear, and how safety standards will evolve in relation to IAEA design requirements. Events such as large-scale sodium fires were originally taken as outside the design basis for LMFBRs based on probabilistic arguments. Some whole core accidents are taken to be within the design basis, thus requiring a higher integrity primary circuit.

The design basis for gas-cooled high-temperature reactors and how it should be interpreted in the light of IAEA design requirements is also as issue.

In future vendors will in practice demonstrate that their design meets IAEA standards (and Utility Design Standards) in additional to national requirements. In the US, the approach of design certification has been used for evolutionary plant. In principle, if a design is approved by the licensing authority in one country, this should at a minimum facilitate its acceptance in other countries. This is based on the assumption that regulators adopt common licensing approaches. A harmonised licensing regime would certainly be beneficial but this is not the present position even in Europe.

Some of the future designs under consideration in the GIF collaboration are not likely to come into commercial operation for many decades. It is probably premature to speculate on the licensing frameworks within which these designs will be licensed. However, it seems likely that any commercial designs that are eventually put forward for licensing will have been developed within increasingly global collaborations and will be licensed against more harmonised international safety legislation.

REFERENCES

Ang, M., Grindon, E., Serro, C., Routamo, T., Auglaire, M., Gustavsson, V., Spanier, L., Ehrstedt, T., Sonnenkalb M., Ewig, F., Droulas, J.-L., Duspiva, J. & Kostka, P. (2001) Optimisation of severe accident management strategies for the control of radiological releases – OPTSAM, FISA 2001: EU research in reactor safety, EUR 20281.

Crech, J. (1999) Engineering challenges of evolutionary reactors: an European vendor prospective on community research in reactor technology, FISA 99: EU research in reactor safety, EUR 19532 EN.

EPRI NP-6780 (1990) Advanced Light Water Reactor – Utility Requirements Document, V.1, Rep. EPRI NP-6780, Palo Alto, CA, 1990.

EUR 16801 EN, ISSN 1018-5 (1996) European commission – nuclear science and technology, licensing procedures and associated documentation.

EUR 20055 EN (2001) 25 Years of community activities towards harmonisation of nuclear safety criteria and requirements – achievements and prospects, October 2001.

European Commission (1975) Council Resolution of 22 July 1975 (77/C 185/1), July 1975.

European Commission (1992) Council Resolution of 18 June 1992 (92/C 172/02), June 1992.

Federal Nuclear and Radiation Safety Authority of Russia (1993) Regulation, nuclear power plant siting, PNAE G-03-33-93, Moscow.

Generation IV Nuclear Energy Systems (2003) http://gen-iv.ne.doe.gov/content.html

Govaerts, P. (1996) Nuclear Regulators Working Group (97/P05), Implementation of the 25 principles of nuclear safety in different EU member states, B4-3070/95-000291/MAR/C2.

Harbison, S.A. (1992) *Safety Assessment Principles for Nuclear Power Stations*, Revised 1992, HMSO, UK.

HSE (1992) *The Tolerability of Risk from Nuclear Power Stations*, Revised 1992, HMSO, UK.

IAEA-TECDOC-905 (1996) Approaches to the safety of future nuclear power plants, Report of a Technical Committee Meeting Held in Vienna, 29 May–2 June 1995, September 1996.

IAEA (2000a) Nuclear technology review 2000, GC(44)9, 14 August 2000.

IAEA (2000b) The agency's programme and budget for 2001, GC(44)6.

IAEA Safety Standard Series (2000) Safety of nuclear power plants: design and safety requirements, NS-R-1, Vienna.

IAEA (2001) Medium term strategy, IAEA web site http://www.iaea.org, March 2001.

IAEA Safety Standards Series No. GS-R-2 (2002) Safety requirements on preparedness and response to a nuclear or radiological emergency.

IAEA/NSR/2002 (2003) Nuclear Safety Review for the Year 2002, International Atomic Energy Agency, August 2003.

IAEA Technology (2002) Annual Report 2002.

IEA/NEA/IAEA (2002) Innovative nuclear reactor development, three-agency study, opportunities for international co-operation.

Ion, S.E., Bull, A.J., Ivens, R. & Abram, T.J. (2003) Meeting Europe's energy needs through next generation nuclear energy systems – an industrial view of the role of EC co-sponsored research, FISA 2003, November 2003.

International Conference on the Safety of Transport of Radioactive Material, Vienna, 7–11 July 2003.

OECD/NEA (1999) The strategic plan of the nuclear energy agency, http://www.nea.fr

OECD/NEA (2000) Annual Report 1999, ISBN 92-64-18504-6, OECD 80823, http://www.nea.fr

USNRC, 10 CFR Part 50 (1988) Emergency core cooling systems; revisions to acceptance criteria, US Federal Register, vol. 53, No. 180, September 1988.

USNRC, 10 CFR Part 52 (1989) Early site permits, standard design certification and combined licences for nuclear power reactors.

USNRC Review of Electric Power Research Institutes (1992–1994) Advanced Light Water Reactor Utility Requirement Document, NUREG-1242, V.1, 2, 3.

Wahlstrom, B. (2003) Risk informed approaches for plant life management: regulatory and industry perspectives, FISA 2003: EU research in reactor safety 10–13.

Chapter 9

Global Developments

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Chapter 9

Global Developments

9.1. INTRODUCTION/OBJECTIVES

This chapter discusses the status of nuclear programmes that are proceeding in the various countries that currently operate nuclear plant. It covers the European countries, North America (US and Canada), the countries of Asia (Japan, Korea, China, and India), the Russian Federation and other areas (e.g. South Africa and Latin America). The majority of countries with nuclear programmes are focussing on water-cooled systems to provide their requirement. Reference is also made to progress with other reactor systems in the countries where they occur. The emphasis though in this chapter is on current and near-term activities; longer term initiatives are reviewed later in the book.

At present, there are significant differences among the countries with current nuclear programmes in regard to their position on nuclear power for the future. New build is continuing in Asia, including some evolutionary plants. One European country, Finland seems likely to place an order for a large water reactor in the near future. In contrast, some countries with large current programmes have moratoria on the building of new plant; others remain uncommitted or neutral. There are also some countries in Central and Eastern Europe that have plants at different (in some cases advanced) stages of completion. Progress has been halted in some cases due to economic or other reasons.

9.2. WESTERN EUROPE

9.2.1 Belgium

There are currently 7 power reactor units operating in Belgium, Doel 1-4, and Tihange 1-3. These are all PWRs. In 2002, these plants produced 60% of domestic electricity, an increase of 2.3% compared with 2001. Although these plants continue to operate at the present time, Belgium announced a moratorium in 1999 on building new nuclear plants and a law was passed in 2002 providing for nuclear phase-out from 2015 onwards (European Commission, 2000; Foratom e-Bulletin, 2003a). This corresponds to reaching the 40-year limit on the operating lifetimes of the plants.

9.2.2 Finland

Finland operates four nuclear power reactors that generate about one-third of the country’s electricity (27% in 2003) (World Nuclear Association, 2003). The plants currently operating are Swedish boiling water reactors, Olkiluoto 1 & 2, operated by TVO, and 2 Russian-designed VVER plants Loviisa 1 & 2 operated by Fortum.

The building of a fifth reactor was approved by the Finnish parliament in May 2002. This is a significant development within Western Europe, because it is the first decision for new build in over a decade. The intention is for the new plant, expected to be the first European pressurised water reactor (EPR), Table 9.1, to be in operation by 2009.

The application to build a new reactor has taken into account economic factors, security of energy supply, and environmental considerations. The economic criteria related to lowest electricity cost, various studies showed nuclear as the cheapest option. The significantly higher capital costs of building and initial fuel load were about three times that of a gas plant but the fuel costs are very much lower. Comparative costs of the nuclear, coal and natural gas options were estimated at 2.40, 3.18 and 3.21 EUR c per kWh on the basis of a 91% capacity factor, 5% interest rate and a 40-year plant life (Foratom e-Bulletin, 2003a), showing the nuclear option to be economically favourable. The nuclear option also showed the lowest sensitivity to possible fuel price increases.

The decision is consistent with the Finnish 1997 energy policy, which stressed availability, diversity of provider, price and security criteria for new energy generation. It also stressed the need to meet international commitments.

9.2.3 France

In France, a first national energy debate is in progress to review energy options over the next few decades up to 2030 and beyond (International Atomic Energy Agency, 2002). It is anticipated that proposals for draft legislation will be put forward towards the end of 2003. This will include definition of the various options for sustainable development at both a national and international level. Along with other countries in Europe, France is seeking a diverse mix of energy generators, including renewables, energy saving and reduction of fossil fuels and careful consideration of the nuclear option. A long-term goal is to achieve a 25% cut in greenhouse emissions by 2050. Nuclear options will include the future of the EPR.

It is well known that France has a very large nuclear energy programme. In 2002, 78% of the electricity share was nuclear power with 59 units operating on over 20 sites (International Atomic Energy Agency, 2002). Each of these sites is being equipped with the latest in simulator technology, to enable improved training and hands-on experience to be gained by the operators.

Table 9.1. Nuclear reactor in Finland

Location/units	Reactor type	Capacity (MWe)	Start of construction	Start up
Olkiluoto 3	EPR	1600	2005 (pending licensing approval)	2009

NIA Industry Link (2004).

Regulatory permission was given in 2003 for the restart of the Phénix prototype fast breeder reactor (FBR) (Foratom e-Bulletin, 2003e). This reactor has been shut for 5 years, following extensive inspection safety up-grading and repair. This reactor is due to close in 2008.

Transmutation is being developed in France to provide a solution to the radioactive waste storage problem by reducing the lifetime and toxicity of long-term radionuclides. The restart of the Phénix reactor will enable 12 transmutation projects to be completed before the closure of the reactor.

9.2.4 Germany

Germany has 13 pressurised water reactors and 6 boiling water reactors currently in operation, contributing to about one-third of the country's electricity generation (World Nuclear Association, 2003). Following the reunification of Germany in 1990, all the Russian-designed VVER plants in the East were shutdown for safety reasons.

Following the formation of a coalition government in 1998 between the Social Democratic Party and the Green Party, it was agreed by both parties to introduce legislation to eventually phase out nuclear power. However, a consensus was agreed between the utilities and the government in mid-2000, which would allow the continued operation of the nuclear plants for some years ahead. There was also a government commitment to allow present reprocessing practices and waste disposal operations to continue. In particular, this allowed for reprocessing in France and the UK and the maintenance of two repository projects in Germany.

In mid-2001, an agreement was eventually signed between the energy companies and the coalition government that limited the operational lives of the reactors to an average of 32 years. In practice, some of the less economic plants are likely to be shutdown sooner. The construction of any new nuclear power plants however remains prohibited at present. An additional principle in the agreement is the storage of fuel on-site.

There is some evidence that German public opinion has moved more towards supporting nuclear energy. It remains unclear whether the country's goals for greenhouse emissions can be achieved without nuclear energy. There is still strong support for Franco-German co-operation in some areas, e.g. in the development of the EPR and in securing the improved safety of Russian-designed reactors via technology transfer.

9.2.5 Netherlands

The Netherlands' sole remaining nuclear plant, Borssele, continues to operate (Foratom e-Bulletin, 2003c) but in 2002, it only contributed to 4% of the electricity generation (International Atomic Energy Agency, 2002). In 1994, the Netherlands declared a moratorium on the building of new nuclear power reactors (European Commission, 2000).

The Dodewaard BWR was closed for economic reasons in 1997, and the last remaining spent fuel assemblies have been shipped to Sellafield in the UK. The plant site will be decommissioned with the intention of returning to a green field site after 40 years.

The Netherlands is however looking to ensure continuation of medical isotopes production after the Petten research reactor reaches the end of its operational life in 2015. The intention is to site a new research reactor at Petten (Foratom e-Bulletin, 2003b).

9.2.6 Spain

There are nine nuclear reactors currently operating in Spain. In 2002, 26% of the country's electricity was generated by nuclear energy. The previous year the figure was 27%; during this period the electricity consumption grew overall by 2.7% (Foratom e-Bulletin, 2003b). In 2002, the average load factor was in excess of 90%.

However, Spain announced a moratorium against building of new nuclear plant as early as 1984 (European Commission, 2000). Currently there continue to be no plans for building a new nuclear plant in the near future.

9.2.7 Sweden

Sweden currently has 11 nuclear power reactors operating, producing about a half of the country's electricity (World Nuclear Association, 2003). In 1980, a referendum was called to examine different options for phasing out nuclear energy. It was decided to continue the operation of existing plants and to complete those under construction provided that it remained economic to do so. The anticipated time period was assumed to be for 25 years, the end of their planned operating lives. At the time, the Swedish Parliament decided against any further expansion of nuclear power with an aim of decommissioning all reactors by 2010.

There had been political manoeuvrings over the last few decades to close Barseback 1 and 2. These are several 600 MWe BWRs operating within about 30 km of Copenhagen and therefore close to the Danish border. In 1997, an agreement was forged between the various political parties to close one unit by mid-1998 and the other unit by mid-2001. In return, the remaining 10 reactors might be allowed to run for 40 years. In practice, unit one was closed in 1999 but unit 2 continues in operation.

Public opinion has been largely supportive to nuclear energy. In a 2001 poll, 75% of people gave the restriction of greenhouse gas emissions as the top environmental priority, only 10% voted for phasing out of nuclear power. On nuclear power matters in general about 76% voted for some degree of nuclear power continuation in Sweden.

Environmental quality is of very high importance in Sweden with commitments to stabilise carbon dioxide emissions at 1990 levels by 2000. A full nuclear power phase out would in fact increase carbon dioxide emissions by about 50% above the 1990 level.

With regard to waste management, there has been an intermediate level waste repository near Forsmark since 1988. For high-level waste, there is the CLAB repository at

Oskarshamn that has been operating since 1985. This is a temporary solution; the fuel will be stored under water in an underground rock repository for about 40 years. It will then be encapsulated in canisters for burial in a 500 m deep repository. Research is underway at the Aspo Hard Rock Laboratory; candidate repositories are at Oskarshamn and Osthammar, at Forsmark.

9.2.8 Switzerland

Switzerland has five units operating which in 2002 generated 40% of the country's electricity requirement. There has been debate on whether Switzerland should exit from nuclear power but in April 2003 (Foratom e-Bulletin, 2003b), the Swiss parliament approved a new law providing for the nuclear option to be left open. This law extends the rights of Swiss citizens to take decisions on the future use on nuclear energy and also in the licensing of waste repositories. There is a recommendation of both government and parliament to vote against forthcoming referenda, which propose to phase out nuclear energy and to replace nuclear power plants by alternative energy sources over an unspecified period.

9.2.9 UK

In February 2003, the UK government published an Energy White Paper (Energy White Paper, 2003) to define an energy policy looking forward from today to 2020 and beyond as far as 2050. Many of the policies set out in the paper took as their starting point the Energy Review published by the Cabinet Office's Performance and Innovation Unit (now the Strategy Unit) (The Energy Review, 2002) in February 2002 and the White Paper was produced after in-depth analysis of the various options. The review covered all forms of energy requirement, from heating and lighting to transport, industry and communications.

Regarding nuclear power for either electrical or non-electrical generation, a key safety issue concerns the management of nuclear waste. Supporters of nuclear energy argue that the technical problems associated with waste disposal are solved; opponents do not agree. There are other commercial and practical issues such as: capital cost, market price of nuclear electricity and energy, and the risks, including liabilities and availability of an adequate skill base. All these will impact any decision for new build.

By 2020, the existing fleets of UK nuclear power stations will all have almost reached the end of their working lives. The White Paper acknowledged that nuclear power was currently an important source of carbon-free electricity and remains an option for the future. However, it did not propose new build and stated that before any decision to proceed with the building of a new power station, there would need to be the fullest consultation and publication of a White Paper setting down Government's proposals. The arguments for a delay were both on economic grounds and concerning the issue of waste disposal. These considerations are clearly relevant to all nuclear energy products (electrical and non-electrical) in general.

Nuclear power in the UK has in the past been used largely for electricity generation, but some reactor designs are suitable for either co-generation of heat or even dedicated nuclear heating applications. For example UK industry is showing a revived interest in high temperature reactors (HTRs). The UK is keeping abreast of a number of international initiatives, via participation in the Generation IV programme led by USDOE.

For many years, fast reactors have offered the attraction of a sustainable fuel supply based on a uranium–plutonium fuel cycle. There is now a current interest in exploring particular advantages of the fast reactor to consume plutonium, and reduce the stockpile of weapons fuel. Also the fast reactor can be used to irradiate minor actinides and fission products to reduce the toxicity of long-term wastes. Within this framework, the gas-cooled fast reactor (GCFR) has a number of potential advantages to offer. The UK is participating in EC initiatives in this area; e.g. an ongoing review of gas-cooled reactor concepts (Mitchell *et al.*, 2001) within the 5th Framework programme.

The UK is also participating in the EC CAPRA (Consummation accrue de plutonium dans les reacteurs Rapides) project, which aims to utilise existing plutonium stocks arising from the operation of commercial thermal reactors (IAEA-TECDOC-1083, 1999).

Work is currently underway in the UK in the EC CAPRA/CADRA project to evaluate the potential for the transmutation of plutonium and minor actinides in a wide variety of reactor concepts including a GCFR or a HTR system (Smith *et al.*, 2003). Participation in these various gas reactor programmes takes advantage of the UK long-standing experience of gas reactor technology.

The UK is also keeping abreast of other initiatives, including the application of proton particle accelerators in connection with sub-critical reactor systems.

The UK participates in fusion research and collaborative international programmes. During the 1990s, the Joint European Torus (JET) project has made progress in generating significant amounts of energy. For the next generation of Tokamaks, interested nations including the UK will participate in the International Tokamak experimental reactor (ITER) project. This technology is not likely to be available as a viable power generator until beyond 2030.

9.3. NORTH AMERICA

9.3.1 Canada

Canada has 14 operating nuclear power plant units, which in 2002 produced 14% of the country's electricity, compared to 13% of the previous year (Foratom e-Bulletin, 2003b). There are a total of 22 nuclear power units but 8 of these have been shutdown for several years. In 2003, permission has been given to load fuel into Units 3 & 4 of Bruce A nuclear power plant. The Canadian Nuclear Safety Commission (CNSC) has granted permission for restart subject to certain specific requirements (Foratom e-Bulletin, 2003c).

Looking to the future, Canada is participating in the Generation IV International Forum (GIF) to facilitate the R & D for these reactor systems.

The long-term management of nuclear waste is under study. In 2002, the Nuclear Fuel Waste Management Organisation was set up to investigate various concepts (World Nuclear Association, 2003). The main proposal under consideration is to bury the waste in the rock of the Canadian Shield, at depths of 500–1000 m, below the water table.

9.3.2 US

A key feature of the US nuclear power industry over the past decade has been a significant improvement in performance. In 2002, the nuclear share of total electricity generation was 20% with an average net capacity of about 90% (World Nuclear Association, 2003). This has been achieved against a background of deregulation of the US Industry following the Energy Policy Act in 1992.

Over the past decade, overall nuclear generating capacity has not changed markedly. Two new plants Commanche Peak 2 and Watts Bar 1 came on stream but 8 reactors were shutdown. The deficit power was compensated for by increased power ratings in the reactors that continued to operate. Further applications for increases in power are expected over the next few years. Increases in operating efficiency have also been achieved through improved maintenance and reducing the length of refuelling outages, from over 100 days in 1990 to 40 days by 2000.

There has been significant consolidation of ownership of the various nuclear plants. In 1991, there were a total of some 101 individual utilities with interest in operating nuclear plants. By 2002, the largest 12 companies owned 68% of the generating capacity. The major player is now the Exelon conglomerate formed by the merger of the two largest owners of Unicom and PECO Energy, together with 3 Amergen units. Other significant mergers resulted from the merger of Carolina Power & Light and the Florida Progress Corporation and the merger of GPU and First Energy. In addition a joint venture Nuclear Management Company (NMC) was set up to operate a number of plants owned by different utilities, each of which owns a share of the NMC.

There are applications with the USNRC for extension of plant lifetimes from 40 to 60 years. Operating licences have been renewed for an additional 20 years for two units of Calvert Cliffs, three units of Oconee, ANO 1, two units of Edwin Hatch, and two units of Turkey Point, North Anna and Surry. Other licence applications are expected for a large proportion of the remaining currently operating reactors. These life extensions will enhance the economic competitiveness of the nuclear plants.

There appears to be resurgence of interest in the possible building of new plants in the US. To achieve even a modest 3% reduction in carbon emissions, a Department of Energy study envisages that not only licensing renewal of existing plants would be required but also the construction of about 30 large nuclear plants by 2012.

There has been a revival of US government R & D funding for nuclear energy to put the US back into a driving position for nuclear technology. The Nuclear Energy Research Initiatives (NERI) programme has been introduced to reinvigorate nuclear research. The Plant Optimisation programme is another example.

Design certification has been provided by the USNRC for three advanced reactor designs over the past decade. The way is available for these plants to be built in the US subject to site-specific licensing considerations. Other countries are also showing interest in these designs. The US is also spearheading the Generation IV programme.

The US is not reprocessing spent fuel, treating it as high-level waste. At present, utilities store the material on-site until a repository becomes available. In 2001, it was agreed that there were no insurmountable technical problems with a site proposed in Nevada. It was agreed by law in July 2002, that this site should become the country's permanent repository.

9.4. ASIA

9.4.1 China

There are currently 8 nuclear power reactors operating in China and there are a further 3 units under construction (World Nuclear Association, 2003; Table 9.2). They contribute to about 1.4% of the country's electricity requirement. These include Daya Bay 1 & 2 that are standard Framatome PWRs at 944 MWe, which have been in operation since 1994; more recently similar reactors Lingao 1 & 2 started up in 2002.

Qinshan 1 was the first locally designed and constructed plant. This was a medium-scale PWR. More recently Qinshan 2, scaled up from Qinshan 1 entered operation in 2002, to be followed by Qinshan 3, expected in 2003.

Qinshan 4 & 5 are heavy-water reactors based on CANDU 6 technology, each at 665 MWe. These came on stream in 2002 and 2003, respectively.

There are also 2 Russian designed VVER-91 1000 MWe units under construction, Tianwan 1 & 2, under an agreement between China and Russia. These units are scheduled to be in operation by 2004.

More reactors are planned under China's Five-Year Plan (2001–2005). These include a further two 900 MWe units at Lingao and up to six more 1000 MWe plants at Yangjiang.

Table 9.2. Nuclear power reactors in China under construction or ready to start building

Location/units	Reactor type	Capacity (MWe)	Start of construction	Start up
Qinshan 3	PWR	610	1996	2003
Tianwan 1	VVER	950	1999	2004
Tianwan 2	VVER	950	1999	2005

World Nuclear Association (2003).

There are further proposals for two 1000 MWe units for Haiyang, two 1000 MWe units at Hui An, and two 1000 MWe units at Sanmen.

The China National Nuclear Corporation (CNNC) has reported that the Sanmen reactors will be PWRs, there are plans for further reactors but the technologies are yet to be decided. Possibilities include a Chinese standard 3 loop design developed in collaboration with Westinghouse or the Framatome CNP-1000 design.

Uranium resources in China are expected to meet the nuclear programme requirements in the short term. Fuel fabrication and enrichment facilities also exist. However, to meet the Country's objective of being self sufficient in nuclear fuel supply, some additional capacity will be required.

Regarding spent fuel treatment and reprocessing, a closed fuel cycle strategy is the declared objective. Construction of a centralised spent fuel storage facility is in progress at LanZhou Nuclear Fuel Complex. There is also a pilot reprocessing plant under construction to be followed up by a full-scale commercial plant.

China is also studying the feasibility of high-temperature pebble reactors to supply process heat for heavy oil recovery or coal gasification. A 10 MWt plant (HTR-10) was commissioned in 2000. China has also a 65 MWt fast neutron reactor under construction near Beijing, scheduled to achieve criticality by 2005.

9.4.2 India

India has a vibrant nuclear power programme with currently 14 units in operation, 9 under construction, and more new reactors planned (Table 9.3). There are 5 research reactors in operation (World Nuclear Association, 2003). Currently, nuclear power supplies less than 4% of the country's electricity requirement. There is a target to reach 10% in 2005. Capacity factors are now much improved compared with a few years ago, reaching 85% in 2001–2002.

The Tarapur plants are increased capacity plants based on domestic technology and are expected to begin operation in 2004–2005. The other PHWRs will follow later; the Rawatbhata units are scheduled to be in operation by 2007. The design for future PHWRs, the first of these are likely to be Kakrapar 3 & 4, has now been raised to 680 MWe.

Table 9.3. Nuclear power reactors in India under construction or ready to start building

Location/units	Reactor type	Capacity (MWe)	Start of construction	Start up
Tarapur 3 & 4	PHWR	490	2000	2004–2005
Kaiga 3 & 4	PHWR	202	2001	2005–2006
Rawatbhata 5 & 6	PHWR	202	2002	2007
Kudankulam 1 & 2	VVER	950	2002	2007
Kalpakkam PFBR	FBR	500	2002	2010

World Nuclear Association (2003).

Two large VVER-1000s are being built by Russia. The first unit is forecast to be commissioned in 2007.

The construction of a 500 MWe fast breeder reactor is in progress at Kalpakkam. This is contributing to the government's objective to utilise India's abundant thorium resource as a nuclear fuel. The intention is for this reactor to be operating in 2010. This reactor will be fuelled by uranium–plutonium-carbide fuel. The plutonium resource would come from currently existing PHWRs.

The intention is to develop an advanced heavy water reactor (AHWR) thorium cycle based of the following route. Existing PHWRs will burn natural uranium to produce plutonium. Fast-breeder reactors of the type above will then burn plutonium and breed U-233 from thorium. AHWRs will then burn the U-233 with thorium. The first AHWR is due to start construction in 2004.

There are plans to build a mix of reactor types to meet India's requirements. The forward plan is to have a 300 MWe AHWR together with a mix of 500 MWe FBRs, 680 MWe PHWRs and 1000 MWe LWRs by 2020.

India's civil nuclear strategy is to achieve complete independence in the fuel cycle. The country has a fuel fabrication facility at Hyderabad for PHWR and BWR. There are also spent fuel and reprocessing plants at Trombay, Tarapur and Kalpakkam. There is a waste immobilisation plant and storage facility at Tarapur.

Research is in progress in setting up a deep geological repository for high-level wastes.

The Indira Gandhi Centre for Atomic Research at Kalpakkam is working on fast reactor technology development. The Bhabha Atomic Research Centre near Mumbai is working on thorium-based systems. In particular, the Centre is working on the AHWR. In addition, India is also developing accelerator-driven systems for driving sub-critical reactors.

9.4.3 Japan

At present, nuclear energy contributes to about 34% of Japan's electricity (World Nuclear Association, 2003) supplied by 53 operating reactors. Apart from an early Magnox reactor, the main focus has been on BWRs and PWRs. Through the years, these designs have been modified to improve operation and performance; recently more advanced ABWRs and APWRs are available. The Japanese utilities initially purchased designs from US vendors, but later domestic Japanese companies such as Hitachi Co Ltd, Toshiba Co Ltd and Mitsubishi Heavy Industry Co Ltd designed and constructed follow-on plants themselves.

In March 2002, the Japanese government agreed to turn to nuclear energy to achieve Kyoto Protocol emission targets. A 10-year plan has been put forward that calls for an increase of nuclear power generation by about 30%. This will be delivered on the expectation that there will be 9–12 new nuclear plants operating by 2010.

There are already a number of ABWRs in commercial operation. There are currently 3 reactors under construction, Hamaoka 5 (BWR), Higashidori 1 (ABWR) and Shika 2

Table 9.4. Nuclear power reactors in Japan under construction or ready to start building

Location/units	Reactor type	Capacity (MWe)	Start of construction	Start up
Hamaoka 5	BWR	1325	2000	2005
Higashidori 1	ABWR	1067	2000	2005
Shika 2	ABWR	1315	2001	2006

World Nuclear Association (2003).

(ABWR) (Table 9.4). There are plans for a further 3 BWRs and 6 ABWRs. These are large plants all in the 1000–1300 MW range. Construction of another 900 MWe PWR is planned and 2 large 1500 MWe APWRs Tsuruga 3–4 is scheduled to start soon. These APWRs are of simpler design to that of earlier PWRs and combine more active and passive cooling systems. The Westinghouse AP-1000 design is being supported by Mitsubishi.

Japan is also supporting other reactor developments. The Joyo FBR has been operating since 1977 and has supplied valuable technical information to be used in subsequent developments. The Monju was a prototype FBR, which started up in April 1994 but has remained closed since December 1995 due to a sodium leakage. There is an aim to restart.

At the end of 1998, the small 30 MWt High Temperature Engineering Test Reactor (HTTR) was started up. This is a graphite-moderated, helium-cooled reactor incorporating ceramic-coated fuel particles in hexagonal graphite prisms.

Earlier, the Japanese developed an advanced thermal reactor at Fugen, including heavy water as the moderator, with light water cooling in pressure tubes. This was a thermal reactor, which used a full mixed uranium and plutonium oxide fuel. There were plans for a 600 MWe demonstration plant at Ohma, but in 1995 it was decided not to go forward.

Japan has the objective to develop a complete domestic fuel cycle capability but based on imported uranium. The Japan Nuclear Cycle Development Institute operates a uranium refining and conversion plant and also an enrichment plant at Ningyo Toge. Japan Nuclear Fuel Ltd (JNFL) operates an enrichment plant at Rokkasho. On this site, there is a high-level waste interim storage facility, the first of its kind in Japan. There is a major fuel fabrication facility at Tokai-mura. JNC operates a pilot reprocessing plant at Tokai-mura together with storage facilities. Also in operation is a high-level waste vitrification plant at Tokai.

Japan is now focussing on four primary reactor designs for the future; sodium cooled with MOX and metal fuels, helium cooled with nitride and MOX fuels, lead bismuth eutectic cooled with nitride and metal fuels, and supercritical water cooled with MOX fuel. These all involve a closed fuel cycle and various reprocessing routes are under consideration. Japan is participating in the Generation IV initiative.

9.4.4 Pakistan

At the present time, nuclear power supplies only a small contribution to the country's energy requirements, generating 2.9% of the total (World Nuclear Association, 2003). Currently operating there are an old PHWR (125 MWe) supplied by Canada and the Chasnupp PWR (300 MWe) supplied by China. Both these reactors are operating under international safeguards. Pakistan also has a 9 MW research reactor.

There are also plans for another Chinese-designed PWR. This is proposed as a second unit at the Chasnupp nuclear plant at Chashma (Foratom e-Bulletin, 2003b).

9.4.5 South Korea

South Korea currently has 18 reactors operating supplying 39% of the country's electricity World Nuclear Association (2003). The reactors operating are PWRs and PHWRs. The first three units were purchased as turnkey projects; later plants involved local manufacturers. There were various vendors, Combustion Engineering (US), Framatome (France) and AECL (Canada).

In the mid-1980s, Korea embarked on a 10-year plan to standardise the design of its nuclear power plants via a collaboration with Combustion Engineering (now Westinghouse). The exception to this plan was the building of three more AECL CANDU 6 units to add to the earlier Wolsong power plant.

The CE System 80 design was chosen as the standardised design and this evolved into the Korean Standard Nuclear Plant (KSNP) design. In addition to CE System 80 features, it also included many US advanced light-water design requirements. All further 1000 MWe units were of this type. In the late 1990s, an improved KSNP+ programme was started. There are 6 such KSNP or KSNP+ units under construction or on order, Ulchin 5 & 6, Shin Kori 1 & 2, and Shin Wolsong 5 & 6. These are scheduled to start up at various times between 2004 and 2010.

The advanced pressurised reactor (APR)-1400 is a further extension drawing on CE System 80+ design features. The System 80+ was chosen because it has USNRC design certification. The design for APR-1400 was completed in 1999 with enhanced safety and a design life of 60 years. The units scheduled are Shin Kori 3 & 4 and 2 units near Ulchin; these are not scheduled for start-up until 2010–2015. By 2015, nuclear power is expected to supply 45% of requirement (Table 9.5).

Fuel cycle facilities exist within the Korea Atomic Research Institute (KAERI) and the Korea Nuclear Fuel Company (KNFC) to supply PWR and PHWR fuel from uranium imported from Canada, Australia and elsewhere.

A revised waste-management programme came into being in 1998. Spent fuel is stored on the reactor site. The intention is to build a centralised storage facility by 2016. The long-term solution for high-level waste is deep geological disposal. Low- and intermediate-wastes are also stored on the reactor site. For this waste, a central repository is envisaged from 2008. This will allow shallow geological disposal of such waste.

Table 9.5. Nuclear power reactors in South Korea under construction or on order

Location/units	Reactor type	Capacity (MWe)	Start up
Ulchin 5 & 6	PWR	950	2004–2005
Shin Kori 1 & 2	PWR	950	2008–2009
Shin Wolsong 5 & 6	PWR	950	2009–2010
Shin Kori 3 & 4	APR	1350	2010–2011
2 units near Ulchin	APR	1350	2015

World Nuclear Association (2003).

Vitrification is also planned from 2006. In 2003, four sites were selected for detailed examination.

In the longer term, there are various plans for extending nuclear-related opportunities. Plans include the development of liquid metal reactors, the direct use of spent PWR in CANDU reactors (the DUPIC process) and utilisation of research reactors. The HANARO 30 MW research reactor started up in 1995. South Korea is participating in the US Generation IV programme.

9.4.6 Other Asian Countries

9.4.6.1 Bangladesh. Bangladesh currently has one operating research reactor. There are plans to reconsider building a 600 MWe reactor (World Nuclear Association, 2003).

9.4.6.2 Indonesia. There are currently three research reactors in operation in Indonesia (World Nuclear Association, 2003). The potential for nuclear power generation is under review. An original feasibility study recommended that first units totalling 1800 MWe should be commissioned about 2004 but according to World Nuclear Association (2003), nuclear power has been deferred indefinitely.

9.4.6.3 North Korea. In North Korea, there are two partially built units and a research reactor (World Nuclear Association, 2003). A South Korean Standard Nuclear Plant type is also under construction.

9.4.6.4 Philippines. The Philippines have one research reactor but it is not currently operating (World Nuclear Association, 2003).

9.4.6.5 Taiwan. Taiwan currently has six units in operation meeting 22% of its electrical energy requirement. Two further advanced reactor units are being built (World Nuclear Association, 2003).

9.4.6.6 Thailand. Thailand has one research reactor and one reactor under construction (World Nuclear Association, 2003). There are some tentative plans to have a power reactor in operation in the next 10 years. It would be followed by five further units.

9.4.6.7 Vietnam. Vietnam has one research reactor (World Nuclear Association, 2003). The country is studying the viability of nuclear power and possibly installing some nuclear power plant by 2010.

9.5. CENTRAL AND EASTERN EUROPE

9.5.1 Armenia

Armenia has one early VVER-270 unit operable.

9.5.2 Bulgaria

Until recently, Bulgaria had four VVER-440/230 units operating at Kozloduy generating 47% of the country's electricity in 2002 (International Atomic Energy Agency, 2002). Units 1 & 2 closed at the end of December 2002 (Foratom e-Bulletin, 2003a). Units 3 & 4 are continuing in operation, due to a pending government decision on their safety, following the outcome of several international reviews (including European Commission and WANO) (Foratom e-Bulletin, 2003e). Further upgrading of Units 5 & 6 is likely to be carried out (World Nuclear Association, 2003).

There are plans for the construction of the Belene nuclear plant to be resumed in 2004, starting up by 2008. A feasibility study is ongoing in 2003. Results are expected in October 2003 after which time the decision to resume or otherwise will be taken (International Atomic Energy Agency, 2002).

9.5.3 Czech Republic

The Czech Republic has six nuclear power plant units in operation, four at Dukovany (VVER-440/213) and two at Temelin (VVER-1000) (Foratom e-Bulletin, 2003b). In 2002, the nuclear share of electricity generation was 25%, (International Atomic Energy Agency, 2002). Of these, the Temelin units are the newest plant – these underwent commissioning and active tests in 2002.

9.5.4 Hungary

In Hungary, there are four operating VVER-440/213 units operating at Paks. In 2002, they produced 36% of the country's electricity requirement (Foratom e-Bulletin, 2003c). Upgrading projects have been implemented for the Paks plants following recommendations from the G7 countries to improve the safety of all Soviet-designed reactors in Central and Eastern Europe.

9.5.5 Lithuania

A very large share of the country's energy requirement comes from operation of the Ignalina RBMK units 1 & 2. In 2002, these plants generated 80% of domestic electricity

(Foratom e-Bulletin, 2003c). Despite improvements, these plants will be shutdown before the end of their 30-year design lifetime (World Nuclear Association, 2003). Ignalina 1 is due to close by 2005 and the closure date for Ignalina 2 will be set in 2004.

The Lithuanian government has approved the design and eventual construction of an interim spent storage facility to be built at the Ignalina site.

9.5.6 Romania

Romania has only one operational unit at Cernavoda 1, which in 2002 supplied just 10% of the country's electricity requirement (Foratom e-Bulletin, 2003c). This is a 655-MWe AECL designed CANDU 6 reactor, which currently has an operating licence through to April 2005 (Foratom e-Bulletin, 2003a). The operating licence is renewed every 2 years.

The completion and commissioning of a second unit, Cernavoda 2 is scheduled. It is currently about 45% complete and commissioning is expected in 2004 (Foratom e-Bulletin, 2003c; Table 9.6).

9.5.7 Russia

During the 1980s, there was a significant construction programme in progress based on the two reactor systems, RBMK and VVER. There then followed a slowing down of progress while the nuclear establishment culture underwent significant changes.

There are currently about 30 units in operation producing in 2003, about 16% of the country's electricity requirement (Foratom e-Bulletin, 2003b). The average capacity factor was about 72%. There are currently about six power reactors under construction, three VVER-1000, Kalinin 3, Volgodonsk 2, Balakovo 5 & 6; one RBMK, Kursk 5 and one fast breeder reactor, Beloyarsk 4 (Table 9.7). There are about 16 plants planned or on

Table 9.6. Nuclear power reactor in Romania under construction

Location/units	Reactor type	Capacity (MWe)	Commissioning
Cernavoda 2	PHWR	655	2004

Foratom e-Bulletin (2003a).

Table 9.7. Nuclear power reactors in Russia under construction or on order

Location/units	Reactor type	Capacity (MWe)	Start up
Kalinin 3	VVER	950	2004
Kursk 5	RBMK	925	2006
Volgodonsk 2	VVER	950	2007
Balakovo 5 & 6	VVER	950	2008–2010
Beloyarsk 4	FBR	750	2009

Foratom e-Bulletin (2003b).

order, including VVER 1000 and RBMK reactors. There are about five thermal power plants also planned.

There is significant emphasis on the improvement of the operation of the existing plants. This includes the utilisation of better fuels, and improved efficiency. There are new fuels incorporating the use of burnable poisons, gadolinium and erbium and also structural changes to the fuel assemblies. All these changes result in improved fuel management.

The philosophy is now to operate a closed fuel cycle, in which MOX fuel is deployed in RBMK and is being introduced experimentally in some VVERs. There are plans to introduce weapons grade plutonium in MOX fuel in up to seven VVER-1000 reactors from 2008. This plutonium will also be utilised in the fuel for the fast reactor Beloyarsk 3.

There is a general acceptance of the need for Russia to continue to press ahead with nuclear energy. Energy demand is rising at 3% per annum. There is also a policy drive and an economic objective to achieve greater exports. There are a number of guidelines for new plant in regard to power and capital cost per plant, a service life of at least 50 years and an utilisation rate of at least 90%, (World Nuclear Association, 2003). There are plans for the nuclear power share to rise to providing 25% of the country's electricity by 2020.

In addition to nuclear power for electricity generation, there are also plans to finish construction of the country's 8th nuclear-powered icebreaker.

9.5.8 Slovakia

There are six units operating in Slovakia that provide a large share of the country's electricity requirement, 73% in 2002 (Foratom e-Bulletin, 2003c). Two of these are the first generation VVER-440/230 design, Bohunice 1 & 2. These have been extensively refurbished in 2003, including modernised control systems and replacement ECCS. Nevertheless, these units are due to close by 2006 and 2008, respectively (World Nuclear Association, 2003). There are also four second generation VVER-440/213 plants in operation, Bohunice 3 & 4 and the newest plants, Mochovce 1 & 2. Upgrading projects for Mochovce, e.g. replacement of the I & C system have been implemented. A similar upgrade is scheduled for Bohunice 3 & 4 (EUR 20056 EN, 2001).

9.5.9 Slovenia

Slovenia has one operating PWR unit, which in 2002 generated 41% of the country's energy requirement (Foratom e-Bulletin, 2003c). The Krsko nuclear power plant is jointly owned with Croatia. The two countries now have an agreement on the status of the plant and also for the management of radioactive waste. It also provides for a joint decommissioning programme.

9.5.10 Ukraine

There are currently 13 reactors operating in the Ukraine, producing a sizeable fraction of the country's electricity requirement. In 2000 for example, the figure was 45.3%

Table 9.8. Nuclear power reactors in Ukraine approved and under construction

Location/units	Reactor type	Capacity (MWe)	Start up
Khmelnitski 2	VVER	953	2004
Rovno 4	VVER	953	2006

World Nuclear Association (2003).

(at this time Chernobyl 3 was still operating) (World Nuclear Association, 2003). The current reactors operating are of VVER design at Khmelnitski 1, Rovno 1–3, South Ukraine 1–3 and Zaporozhe 1–6, making the Zaporozhe station the largest operating station in Europe.

There are currently five reactors at certain stages under construction. In 2000, the European Bank for Reconstruction and Development approved the completion of Khmelnitski 2 and Rovno 4 (Table 9.8). The scheduled dates for commissioning are 2004 and 2006, respectively. Construction on the remaining three, Khmelnitski 3 & 4 and South Ukraine 4 was stalled indefinitely.

Ukraine is setting up special working groups to consider proposals for management of the country's nuclear waste (Foratom e-Bulletin, 2003b). These will be put forward in 2003. A pioneering method for the treatment of nuclear waste is being developed via collaboration with Russia and France. It was observed after the Chernobyl accident that the substance chitin, occurring naturally in fungi and insects, is an extractor of heavy metals such as uranium and plutonium. It is being investigated whether this process could be applicable to the processing on spent nuclear fuel.

9.6. SOUTH AFRICA

South Africa has 2 PWR units operable in its present nuclear programme.

The South African utility ESKOM is ready to go forward on the development, construction and commissioning of a demonstration unit for the pebble bed modular reactor (PBMR), subject to the required statutory approvals (Foratom e-Bulletin, 2003d).

9.7. LATIN AMERICA

9.7.1 Argentina

Regarding the present, there are two PHWRs operating in Argentina. Regarding the future, Argentina is a member of the Generation IV Forum initiative.

9.7.2 Brazil

Brazil has two PWRs operational at the present time. Brazil has also signed two 'energy' partnerships with the US which amongst other objectives are to aid research and development in advanced nuclear technology. This relates particularly to the international Generation IV initiative (Foratom e-Bulletin, 2003e).

REFERENCES

Energy White Paper (2003) The Licensing Division, HMSO, Norwich, Crown Copyright 2003.

European Commission (2000) Green Paper: Towards a European Strategy for the Security of Energy Supply, November 2000.

EUR 20056 EN (2001) Main characteristics of nuclear power plants in the European union and candidate countries, October 2001.

Foratom e-Bulletin, February 2003a.

Foratom e-Bulletin, April 2003b.

Foratom e-Bulletin, May 2003c.

Foratom e-Bulletin, June 2003d.

Foratom e-Bulletin, July 2003e.

IAEA-TECDOC-1083, Summary and future trends, status of liquid metal-cooled fast reactor technology, April 1999.

International Atomic Energy Agency, 2002.

Mitchell, C.H., Farrar, B., Lo Pinto, P., Dominguez, M.T., van Heek, A., Every, D. & Carluec, B. (2001) Gas-cooled fast reactor concept review studies (GCFR), FISA 2001: EU Research in Reactor Safety, Luxembourg, 12–15 2001, EUR 20281.

NIA Industry Link (2004) Issue No. 3, January/February 2004.

Smith, P.J., Hosking, J.G. & Newton, T.D. (2003) Fuel heterogeneity in a particle fuelled ETGCFR core design, SA/PSS/14339/W003 Issue 1, April 2003.

The Energy Review (2002) A Performance and Innovation Unit Report, Cabinet Office, Admiralty Arch, London, SW1A 2WH, February 2002.

World Nuclear Association (2003) http://www.world-nuclear.org/.

Chapter 10

Evolutionary Water Reactors

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Chapter 10

Evolutionary Water Reactors

10.1. INTRODUCTION/OBJECTIVES

The purpose of this chapter is to describe briefly the evolutionary water reactor designs that have evolved from current generation commercial reactors. These evolutionary designs have been developed during the 1990s, taking advantage of lessons learned from existing plant. The chapter focuses on water reactor systems because these occupy the dominant position among the evolutionary reactor designs that are currently under consideration for building in the short term. Other types of advanced reactors are considered later in the book. There is no attempt to describe all possible designs in detail. Rather the approach is to categorise the various designs into different types and then describe the representative features of the reactors within a given type. This enables the reader to understand the general design features that are currently being put forward. References are given for the comprehensive range of reactor types.

Various evolutionary improvements have been proposed for all the major water reactor types currently in operation, i.e. PWRs, BWRs, and HWRs. Common general features are simplification in design to reduce cost, coupled with increased safety features. Many of the designs are available at different power capacity ratings, from medium size, e.g. ~500–600 MWe range, through to 1000–1300 MWe range. These have been put forward to provide more flexibility to meet the current market demand but also have evolved to meet perceived changes in demand. There was a trend in the mid-1990s to produce medium-range designs to take advantage of increased passivity in design. However, the economics of larger plants are now thought to be more favourable, and present trends are more towards the larger plant scale. Further it has been shown that the medium-sized passive designs can be scaled up.

10.2. LIGHT WATER REACTORS

Table 10.1 lists some of the large evolutionary designs that have been put forward during the 1990s.

Table 10.2 lists a selection of the medium evolutionary designs that have been put forward in the 1990s.

10.2.1 Pressurised Water Reactors

10.2.1.1 EPR. The European pressurised water reactor (EPR) was initiated by Siemens and Framatome and their subsidiary, Nuclear Power International (NPI)

Table 10.1. Advanced large evolutionary reactors

Reactor	Design organisation	Capacity (MWe)
PWR		
EPR	Nuclear Power Int.	1750
APWR	Mitsubishi/Westinghouse	1530
System 80 +	ABB Combustion Eng.	1350
KNGR	Kepco/Korean Ind.	1350
AP1000	Westinghouse	1000
EP1000	Westinghouse/Genesi	1000
BWR		
BWR90/90 +	ABB Atom	1190-1374/1500
ABWR	Hitachi/Toshiba/GE	1315
ESBWR	GE	1190
SWR 1000	Siemens	1000
VVER		
VVER-1000 (V-392)	Atomenergoproject	1000
HWR		
CANDU 9	AECL	935

Data from IAEA-TECDOC-1117 (1999) and IAEA-TECDOC-968 (1997).

(Fischer *et al.*, 1999). The venture was carried forward with the co-operation of the major German Utilities and Electricité de France. It took account of the earlier proven German KONVOI and the French N4 designs and the operating experience of previous Siemens and Framatome designed plants.

The initial EPR design was finalised by 1997 and a preliminary safety analysis report was published. Following this, a design optimisation phase was carried out and the maximum design power increased to 1750 MWe to achieve more competitive economics.

Table 10.2. Advanced medium evolutionary reactors

Reactor	Design organisation	Capacity (MWe)
PWR		
AP600	Westinghouse	600
AC-600	NPIC	610
BWR		
SBWR	GE	670
HSBWR	Hitachi	600
VVER		
VVER-640 (V-407)	Atomenergoproject	640
HWR		
CANDU 6(E)	AECL	700
AHWR	BARC	235

Data from IAEA-TECDOC-1117 (1999) and IAEA-TECDOC-968 (1997).

The design aimed to comply with common requirements of the German and French licensing authorities, with the intention that the plant would be licensable in both countries. Further, EPR development was carried out to be consistent with EURs under development at the time. In addition to safety, the aim was to achieve competitiveness of nuclear power against alternative energy options.

The main objective of EPR was to simplify the safety systems, including the elimination of common failure modes via the inclusion of proven active safety systems and diverse back-up systems chosen to be consistent with an evolutionary approach. Increased grace periods for operator actions were achieved by designing components with larger water inventories and reduced sensitivity to human errors.

Particular attention was paid to improving severe accident defence within the design. Severe accident frequency has been reduced by deterministic design criteria and verified using probabilistic verification of the design. The consequences of severe accidents have also been limited via some specific design features to reduce the consequences of core melt scenarios. A compartment is provided with a protective layer and with a provision for active cooling of the base-mat. Passive flooding of the compartment with water is allowed for after corium spreading. The specific design features are designed to protect molten core–concrete interaction.

The measures to ensure economic competitiveness of the EPR include a goal of plant availability of 92%, efficiency of 36%, reduced building costs, design lifetime of 60 years, reduction of fuel cycle costs aiming for burn-ups in excess of 60 MWd kg^{-1} and large unit power capacity of 4900 MWt.

Figure 10.1 is a schematic of the EPR containment showing some of the evolutionary enhanced safety design features of EPR, including the in-containment refuelling water storage tank (IRWST) and the large compartment area beneath the reactor vessel.

10.2.1.2 APWR. The advanced pressurised water reactor (APWR) was designed via a collaboration between the Japanese utilities Ihokkaido, Kansai, Shikoku, Kyushi Electric Power Companies and the Japan Atomic Power Company, organised by the Japanese Ministry of Trade and Industry, and the Westinghouse Electric Corporation (Yamaguchi *et al.*, 1999). This design was evolutionary; taking account of construction and operating experience of PWRs, including that gained from the 23 PWRs currently in operation in Japan.

The main design features are a core design to improve the effective use of uranium fuel, enhancement of the reliability in the reactor internal structures and improved safety via an improvement of the engineered safeguard systems.

An improved radial reflector was introduced to improve neutron economy and also to reduce the fluence to the reactor vessel and internals. This is an important feature in facilitating the proposed 60-year design lifetime.

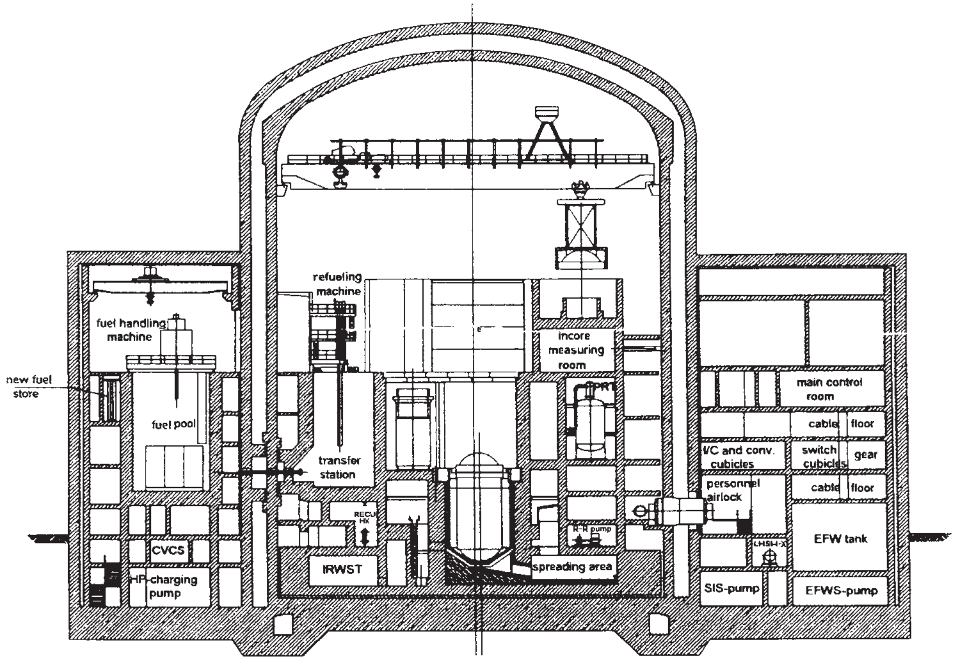

Figure 10.1. European pressurised reactor (EPR). Source: Meyer (1992).

Another evolutionary feature is the introduction of an advanced accumulator. This is a passive device pressurised by nitrogen gas to enhance reliability. These accumulators inject at high rate during the early stage of a primary circuit depressurisation accident, e.g. LOCA, in the same way as conventional accumulators. However, the injection flow is then reduced to a lower rate, allowing the elimination of low head injection. This results in a simplification and, therefore reduced cost.

To achieve economies of scale, the power level is now proposed at 1530 MWe. This is enabled through design optimisation (the original design power was 1420 MWe), specifically via an efficiency improvement of the steam turbine and the reactor coolant pumps, without further change required to the main system configuration or components.

10.2.1.3 System 80 +. The System 80 + design (Matzie and Ritterbusch, 1999) has been developed by ABB Combustion Engineering Nuclear Power based on an evolutionary process that aims to address both improved economic and safety objectives. This approach has evolved from the System 80 plant, e.g. implemented at the Palo Verde NPP and the Korean Standard NPP (KSNPP) designs. The System 80 + Standard Plant Design has been designed to meet the EPRI URD and was certified by the USNRC in 1997. The power rating is currently at 1350 MWe.

The design incorporates a number of features to increase redundancy, diversity and simplification.

The emergency feed water system (EFWS) has two components, each with two emergency feedwater pumps and one EFWS storage tank, giving a 100% increase in EFWS pump redundancy, together with other features to increase diversity and simplification.

The safety depressurisation system (SDS) vents steam from the pressuriser to allow feed and bleed following a total loss of feedwater or prevent high-pressure melt ejection under severe accident conditions. It consists of redundant piping trains to sparge into the IRWST.

The safety injection system (SIS) has four high-pressure pumps from the IRWST and four medium-pressure tanks, which are pressurised with nitrogen and inject water via passive injection. As for other advanced designed designs, e.g. APWR, low-pressure injection pumps are not required.

Other features include the containment spray system (CSS) providing water to cool molten fuel under severe accident conditions. It consists of two components, each with two pumps and two heat exchangers (HXs). This takes water from the IRWST. This gives 100% increase in cooling water pump and heat removal capacity.

Finally, the cavity flooding system is in place to mitigate ablation of the cavity concrete by molten core under severe accident conditions.

Most of these improvements have been incorporated into the KNGR design, described next.

10.2.1.4 KNGR. The Korean next generation reactor (Kim and Kim, 1999) is under development by KEPCO and the Korean nuclear industry. It is an advanced development of the Korean standard nuclear power plant (KSNP) design. The development project has progressed through four phases, starting in the early 1990s with the objective of the first unit becoming operational on the grid by 2010.

The KNGR design is consistent with the requirements of the Korean utility requirements documents (KURD). It incorporates advanced features enabling operation at increased power, enhanced safety, increased margins, improved operation and maintenance characteristics, lower cost and longer design life.

The power output of KNGR is about 40% higher than the KNSP; the operating power on KNGR is intended to be 1300 MWe. Increased margin and lower core outlet temperature has been achieved in KNGR (compared with KNSP).

Safety system reliability and simplicity has been achieved by having four independent mechanical trains. Injection of the SIS is directly into the vessel downcomer, which is less vulnerable to cold leg break accidents inherent in the cold leg injection design of the previous design. This system takes suction from an IRWST that completely surrounds the reactor cavity.

An integrated reactor head assembly has been adopted for KNGR, in place of the multi-component reactor head design of the KSNP. The new design results in reduced refuelling outage duration and also reduced radiation exposure for operating staff.

Various improvements have been made to the reactor coolant systems (RCSs) components, to enable increased power operation for a 60-year plant life and 90% availability. These include increased pressuriser volume, to better accommodate transients and reduce loads on the plant safety system, and the incorporation of Inconel 690 tubes in the steam generator (SG) to reduce stress corrosion cracking.

10.2.1.5 AP1000. The AP1000 has been developed by Westinghouse as an extension of the AP600 design, described later. It is a two-loop 1000 MWe pressurised reactor, with only minimal changes to the AP600 design (AP1000, 2002) (Figure 10.2).

The main changes relate to component changes to accommodate the increased rating while retaining the safety margins. These include increase in the steam generator transfer area and increased coolant pump size. The larger pumps provide increased inertia of the flywheel and hence improved safety margin for pump trip transients. The containment is also higher (but not wider) in view of the increased mass and energy present in the reactor system.

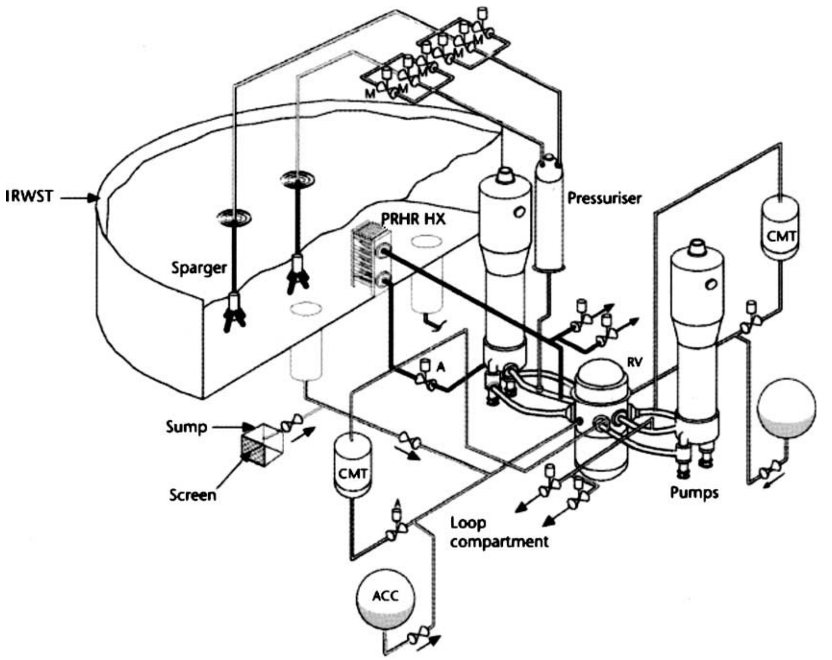

Figure 10.2. AP1000. Source: AP1000 (2002).

Safety assessments carried out for AP600 and AP1000 show that the safety characteristics are not power level dependent. This has been demonstrated by probabilistic risk assessments (PRAs) for the respective power ratings.

Particular attention has been paid to achieving improved construction targets compared with historical experience. The target for AP1000 is a construction schedule of 36 months from first concrete to the load of fuel. This can be achieved through the design simplifications in AP600 and AP1000 resulting in fewer components to install. Another feature is the modular design. Many modules can be built in parallel with other site activities.

As with all credible advanced plant proposals the economics of generation must be competitive. The total generation costs are below \$36 per MWh for a standard twin AP1000 unit. Capital costs are also competitive, estimated at \$1100–\$1200 per kWe for the over-night capital cost.

The AP1000 design is currently under review by the USNRC for design certification.

10.2.1.6 AP600. The Westinghouse AP600 design has been developed as part of the co-operative advanced light water reactor (ALWR) programme supported by the USDOE and EPRI. The design has been developed to satisfy DOE standards and also the ALWR URDs (Ganglaff, 1999).

A key design objective was to design a simplified ALWR which meets USNRC regulatory requirements, compliant with the above standards, yet remaining economically competitive compared with other power systems.

The AP600 safety systems are almost all passive, reliant upon gravity, natural circulation, natural convection, evaporation and condensation in place of other power systems. There are three important safety systems, including passive residual heat removal (PRHR), passive safety injection and passive containment cooling.

The PRHR system includes a HX which protects the plant against transients. The PRHR HX consists of a bank of tubes connected to the RCS, via a circuit, normally isolated from the RCS but with valves that fail open if power is lost. Heat is then removed from the RCS by natural circulation in transients. There is sufficient heat capacity for about 2 h before boiling commences; ensuring steam is condensed in the containment before being returned to the IRWST.

The passive SIS utilises three sources of water including core make-up tanks (CMTs), accumulators and the IRWST. The CMTs accommodate small leaks at any RCS pressure using only gravity. High-pressure accumulators fulfil the make-up function for LBLOCAs. Long-term injection is provided for by the IRWST.

The passive containment system transfers heat from the containment to the ultimate heat sink. Steam condenses on the inner steel liner of the containment. The outside is cooled by a flow of a natural circulation and by water evaporation from sprays on the exterior. The water tank providing the water has sufficient inventory for three days.

The AP600 design has been assessed against severe accidents using PRA methods. The core melt frequency is estimated at 1.7×10^{-7} per year compared with the ALWR target of 1.0×10^{-5} per year. There is a capability to flood the cavity to provide vessel cooling and prevent vessel failure. There is an automatic depressurisation system (ADS) which eliminates the threat of high-pressure core melt injection. There is also an ignitor system to mitigate hydrogen under severe accident conditions.

The AP600 project has been supported by an experimental programme to confirm the design philosophy and validate the design computer codes. This is described later in the book.

Westinghouse received design certification from the USNRC in September 1998.

10.2.1.7 AC-600. This is a passive advanced PWR under development by the Nuclear Power Institute of China (NPIC) and is based on the 610-MWe Qinshan design (Huang and Zhang, 1999). Principal features include an 18 ~ 24 month fuel cycle, low core power density with no vessel penetrations below the primary coolant elevations. It takes advantage of the proven Qinshan technology. Simplified systems and reduced number of components are a design objective.

It includes a number of passive safety systems, a passive emergency residual heat removal system, passive and active SISs and passive containment cooling. It also incorporates an advanced control room layout. The overall design is based on modern international design trends.

10.2.2 Boiling Water Reactors

10.2.2.1 BWR 90. The BWR 90 boiling water reactor design has been developed by ABB Atom AB and is based on experience with the advanced BWR 75 design. Modifications have been introduced to take advantage of technological progress, more stringent safety requirements and to achieve cost savings (Haukeland *et al*., 1999).

In relation to the reference design, the changes are relatively minor. The main design features are internal recirculation pumps; such pumps have operated reliably in ABB BWR plants since 1978. The engineered safety systems are divided into four redundant subsystems which are physically separated (only two are required in the design basis), also based on the BWR 75 concept. The control rod drives system is based on the traditional ABB BWR approach.

Some components have been modified slightly to achieve cost savings, the number and length of welds have been reduced in the reactor pressure vessel, via large section forging. Building volumes have been reduced. Simplification of power systems is another feature.

The BWR 90 containment has been strengthened to mitigate possible effects of core melt severe accidents. Thus severe accident protection capabilities have been enhanced.

By taking advantage of various new technical developments, it has been possible to upgrade the power to 1374 MWe, compared with the reference plant utility requirements. The BWR 90 design has been reviewed against the European Utility Requirements.

10.2.2.2 BWR 90+. A new evolutionary design is also being developed to offer reduced costs and significant safety improvements. The development work is being carried out in collaboration with TVO. The design target is for a 1500 MWe power rating with a build time of 1500 days (Haukeland *et al*., 1999), 90% availability and a refuelling outage of 15–20 days per year.

The BWR 90+ design, builds on proven designs, and is based on established international codes and standards. It takes account not only of EUR but also of the EPRI URs. In addition, particular attention is paid to recent STUK guides and the needs for improved accident mitigation and limited consequences.

The containment has been substantially improved, taking advantage of modular building techniques to reduce construction time and therefore costs. Pipe connections between the drywell and wetwell have been eliminated, except for vacuum breakers, to minimise any potential for drywell–wetwell bypass. Core cover is maintained if a LOCA were to occur during refuelling. A core catcher is arranged under the pressure vessel, which in the case of a severe accident would collect molten corium and then be cooled by water. This also helps to reduce the probability for steam explosions and reduces the risk of core–concrete interaction. Other features include, increased volumes to prevent pressure build-up from hydrogen in a severe accident challenge, and nitrogen gas inerting to allow sprays to operate without risk of hydrogen explosions, and filtered containment venting.

There are further plans to improve decay heat removal by incorporating a passive heat removal system based on the isolation condenser system in early ABB designs.

10.2.2.3 ABWR. The advanced boiling water reactor (ABWR) was developed by General Electric (GE), Hitachi, Toshiba, Japanese Utilities and Government (Nishimura *et al*., 1999) to meet the requirements of a high-performance BWR.

New features included a reactor internal pump for the recirculation system to replace the conventional external loop system. A fine motion control rod drive replaced the conventional locking piston control drive.

A re-enforced concrete containment vessel was adopted in place of the thick steel containment vessel in the past designs. In the new design, there is a thin steel liner to prevent leakage, while the concrete provides the pressure containment function.

To enhance safety injection capability and redundancy, the new design incorporates an ECCS consisting of three high-pressure injection systems and three low-pressure trains.

A better turbine system thermal efficiency and improved system architecture is adopted, compared with earlier designs.

High-level development goals included core damage frequency less than 10^{-7}, a construction period of less than 48 months, first refuelling outage of less than 55 days and occupational radiation exposure less than 0.36 man-Sv per year.

The ABWR is sized at 1315 MWe and is therefore a large-scale BWR.

Two units at the Kashiwazaki-Kariwa power station (units 6 and 7) come into operation in 1996/97. The ABWR is designed to be compliant with the EPRI URD. The ABWR received design certification from the USNRC in 1997.

10.2.2.4 ESBWR. The European simplified boiling water reactor (ESBWR) is an uprated version of the 670 MWe SBWR of GE described below (IAEA-TECDOC-968, 1997). The design objectives include a 60-year plant life, with a full cycle based on a 24-month refuelling interval. Safety functions are maintained through passive means with 72-h grace time for design basis accidents. The ESBWR uses proven design technology developed from the benefits of many man-years experience of operating large plants. Some innovative features are included resulting in a simple direct cycle plant.

10.2.2.5 SWR 1000. The SWR 1000 concept was developed by the Siemens Power Generation Group (KWU) in collaboration with the German nuclear utilities (Brettschuh, 1999). It is an advanced 1000 MWe BWR plant, aimed at being economic compared with coal-fired stations, and to exhibit a high degree of safety, with increased protection against core melt accidents (Figure 10.3).

Passive safety systems have been introduced, which have no need of I&C equipment or active power systems to operate, functioning under gravity, heat transfer and natural convection.

Emergency cooling condensers provide passive heat removal from the reactor pressure vessel. These condensers are in the core flooding pool and they are connected to the core. Condensate is returned to the reactor pressure vessel via gravity flow.

The plant also contains containment cooling condensers for passive heat removal from the containment. These remove heat from the containment to the dry-separator storage pool above the containment but inside the reactor building, following a loss of active heat removal systems.

There are other passive systems, e.g. pressure pulse transmitters to initiate reactor scram, containment isolation of main steam lines and valve operations.

The design is compliant with German design codes, standards and specifications, and with IAEA guidelines and EURs.

Key safety features include a low power density and large water containing vessels stored above the core inside the reactor pressure vessel, in the pressure suppression pool and elsewhere. The plant is designed such that all accidents can be mitigated by passive safety systems with a grace period of at least three days.

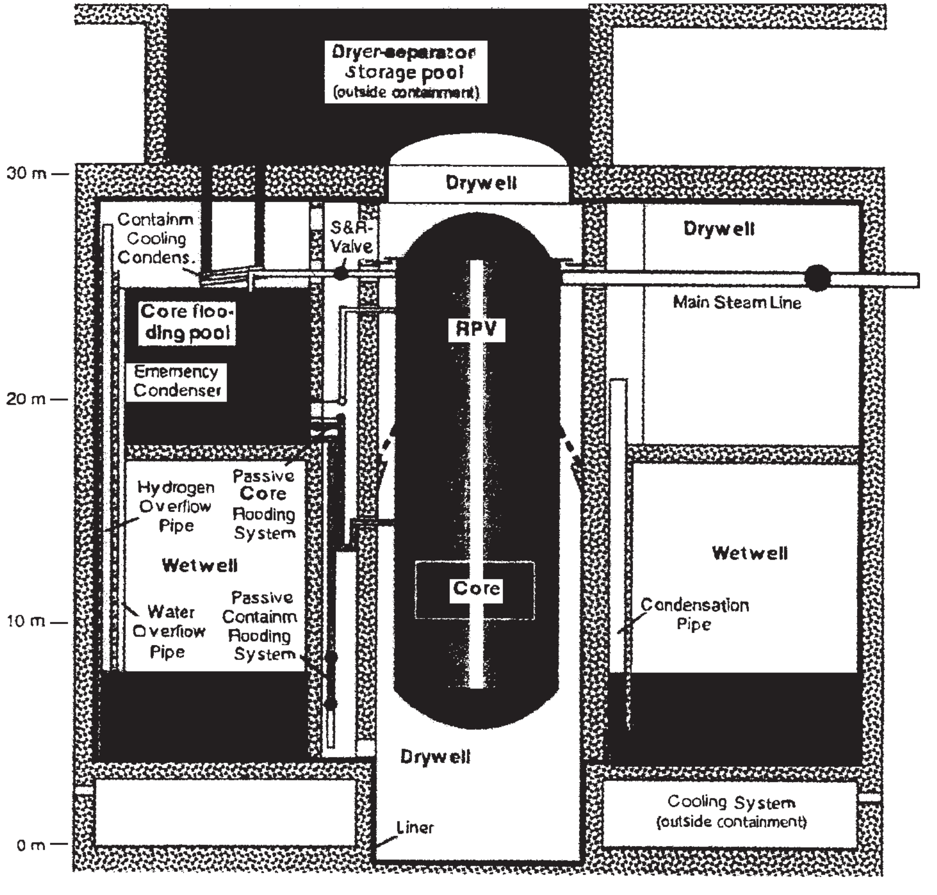

Figure 10.3. SWR 1000. Source: Yadigaroglu *et al.* (1998).

There are a number of features to control core melt accidents. Melt is retained within the reactor pressure vessel, by cooling its exterior. This is achieved by a passive system that supplies water from the core flooding pool to the drywell. This can be activated manually.

10.2.2.6 SBWR. The SBWR design was developed by GE in the mid-1980s with a nominal design of 600 MWe (IAEA-TECDOC-968, 1997). Principal design features include a lower power density than a conventional BWR, which results in improved fuel cycle costs and more manoeuvrability. It operates with natural circulation of primary coolant, and like the ESBWR contains various innovative features in a simple direct cycle plant. Important new design features include an isolation condenser system, a gravity driven cooling system and a passive containment cooling system. Inerting systems are also included in the containment.

10.2.2.7 HSBWR. The Hitachi Simplified BWR has been developed by Hitachi as a medium-size range reactor of 600 MWe (IAEA-TECDOC-968, 1997). It also adopts natural circulation on the primary coolant and passive safety systems. The containment is standardised. Again the emphasis is on economy, good maintenance characteristics and reliability by design simplification.

10.2.3 VVER Systems

10.2.3.1 VVER-1000 (V392). The VVER-1000 (V-392) has been designed by Atomenergoproject/Gidropress. An important objective (IAEA-TECDOC-968, 1997) is to minimise radiation doses through the design at all possible levels of release from normal operation and frequent faults and in design and beyond design basis accidents. It includes design developments to meet severe fuel damage frequencies of 1.0×10^{-6} per reactor year based on a consistent implementation of the defence-in-depth principle. The principal design is underpinned by more than 100 reactor years of VVER-1000 type operation.

10.2.3.2 VVER-640. The VVER-640 reactor (Model V-407) has been developed by O K B Gidropress, taking advantage of the extensive operating experience from previous VVER-440 and VVER-1000 reactors operating in Russia and Central Europe (Dragunov *et al.*). It is designed for a 50–60-year plant life.

The design, therefore takes advantage of proven technologies with increased reliability. The safety systems are passive; the grace period extends to 24 h. It conforms to current regulatory requirements in Russia, themselves designed to be consistent with IAEA guides.

The core exhibits decreased core power density. It was developed from the VVER-1000 design but with this as a design objective. Large water inventories are available in the vessel and the pressuriser.

The passive safety systems are available to supply emergency core cooling under design and beyond design accident conditions. There is a passive heat removal system (PHRS), an ECCS system, and a system for primary circuit depressurisation. The only active system is a high-pressure boron injection system for mitigating anticipated transient without scram (ATWS) accidents.

The PHRS provides heat removal from the containment and SGs. The containment consists of a primary hermetic component made of steel and a secondary component made of concrete. The latter is seismic qualified and also qualified against external hazards such as aircraft crash. The PHRS works by natural circulation to tanks located on the outside of the concrete containment. Heat removal from the SGs takes place under transient conditions. The PHRS works by opening valves on the pipelines connecting the SGs with HXs in the tanks.

Under LOCA conditions, the system is depressurised; this connects the ECCS tanks to the reactor circuit. The ECCS injects water from a hydro-accumulator and by gravity, from the ECCS water tank. Eventually long-term cooling is established by natural circulation through the emergency pool.

The plant also contains systems for mitigating severe accidents. It includes control of hydrogen and reactor cavity flooding to cool the vessel and prevent vessel melt-through under core melt conditions. In the analysis of PRA, the severe core damage probability is now estimated to be in the order of $10^{-9}–10^{-10}$.

The V407 plant has an advanced I&C system, providing improved reliability, integrated diagnostics and expert systems to brief operators on plant conditions.

The Russian regulator, Gosatomnadzor, has issued licences for the Sosnovy Bor site and also for Kola NPP Unit 2 plants.

10.3. HEAVY WATER REACTORS

10.3.1 CANDU Designs

10.3.1.1 CANDU 9. The CANDU 9 reactor has been developed by Atomic Energy of Canada Ltd (AECL) (Yu, 1999). It is based on the multi-unit Darlington and Bruce-B designs including some additional features. The developments are based on proven systems and components. The rating is 935 MWe. The Atomic Energy Control Board (AECB) has confirmed that there are no conceptual barriers to licensing in Canada (Figure 10.4).

Key safety improvements include the following. The containment building is a steel lined pre-stressed concrete structure. The building is based on the 'large dry' concept and there is no need for the dousing system that was needed to achieve enhanced containment integrity in earlier CANDU designs. It also has a lower design leakage rate, enabling reduced siting area requirements.

The system also incorporates more reliable isolation of the ventilation line containment penetrations and, in the event of a severe accident, better hydrogen mitigation via installation of hydrogen ignitors/recombiners.

There have also been improvements to the RCS. Adjacent channels are alternately connected to separate inlet and outlet headers, enabling more even distribution and minimising the positive reactivity insertion in the event of a LBLOCA in the heat transport system (HTS). A larger pressuriser is provided to accommodate volume changes from full power to shutdown.

There is improved layout and separation of the steam and water systems and the electrical systems. This ensures that common mode events do not impair the systems' safety function. The main control room is qualified against seismic events and remains functional for all design basis accidents.

A reserve water system is included to provide better mitigation of severe accidents. This provides emergency water supply for various low pressure cooling loads

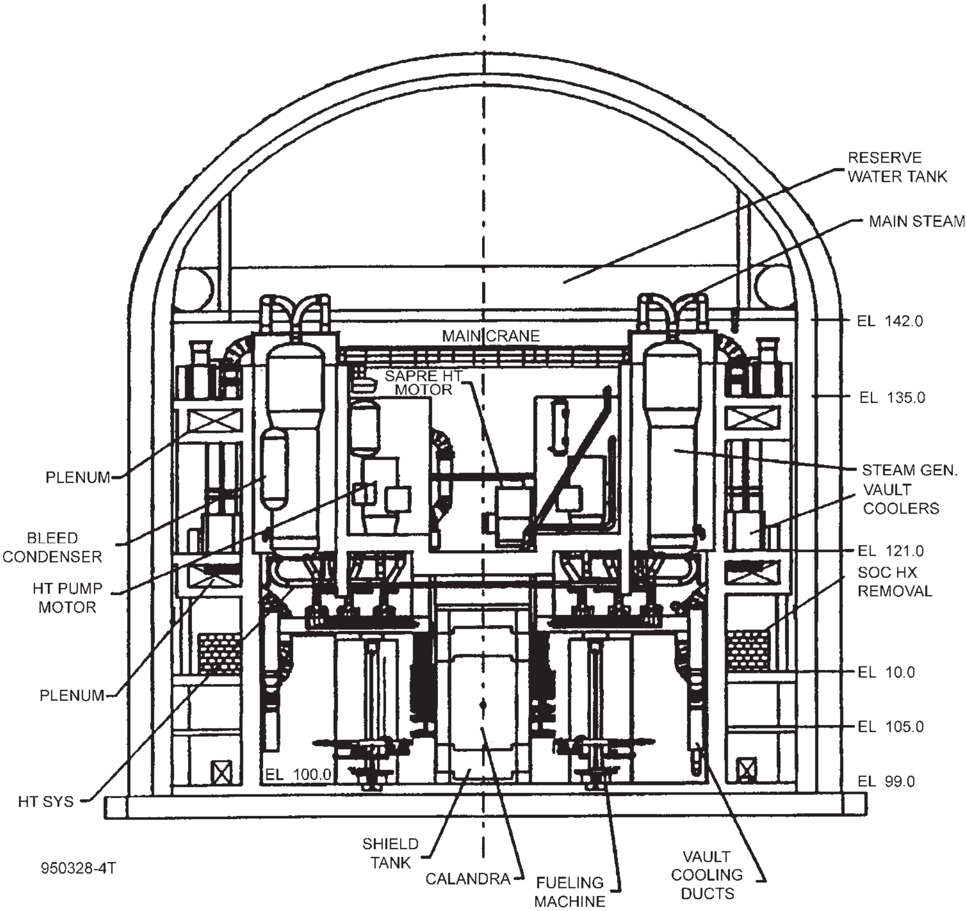

Figure 10.4. CANDU 9. Source: Yadigaroglu *et al.* (1998).

(e.g. low-pressure coolant injection and back-up feed water supply). It also provides a make-up source for the shield tank, moderator and heat transfer systems.

The ECCS has improved reliability and performance by including simple rupture discs and floating-ball isolation valves in place of conventional valves. High-pressure injection valves have been eliminated. Other improvements are the location of the ECC tanks inside the containment and shorter injection lines.

10.3.1.2 CANDU 6. The AECL CANDU 6 nuclear power plants have been operating successfully since the early 1980s (Hopwood, 1999). These are rated at 700 MWe. Since 1996, a development programme has been in progress to improve technical design, decrease commissioning time and improve operating performance. Design upgrades have been incorporated in the latest plants such as Wilsong-4 in Korea and Qinsham 1&2 in China.

Features of the advanced CANDU 6 design include an advanced control centre to improve the human machine interface and other features.

An advanced fuel design based on the CANFLEX fuel bundle has been introduced to allow flexibility for different fuels. For the current fuel, an advanced 43 element fuel bundle enables the same overall bundle power at 20% lower pin rating compared with the earlier 378 elements fuel design.

Incremental improvements have been made to the safety design including increased redundancy of components, simplified containment design and as indicated above, improved fuel thermal margin.

There have been improvements to the power system design, including improved materials for pipes (e.g. higher chromium content) to achieve a 60-year operating life.

Many of the engineering and construction techniques are common to both the CANDU 6 and the CANDU 9. The CANDU systems are summarised in various publications from AECL, see example, AECL.

The Advanced CANDU Reactor (ACR-700) is an evolution of the CANDU 6 design (AECL). The innovations included in the design allow significant cost reductions to be achieved compared with the original CANDU 6 design. The target overnight capital cost for ACR is set at $1000 per kWe.

REFERENCES

AECL, Business Development Office, Ontario, Canada, L5K 1B2.

AP1000: set to compete, *Nucl. Engng Int.*, October 2002.

Brettschuh, W. (1999) SWR 1000: the new boiling water reactor power plant concept, summary, IAEA-TECDOC-1117, December 1999.

Dragunov, Yu.G., Mokhov, V.A., Nikitenko, M.P, Afrov, A.M. (1999) Key developments in the advanced NPP with WWER-640/V-407 reactor plant design, IAEA-TECDOC-1117, December 1999.

Fischer, U., Bouteille, F. & Leverenz, R. (1999) The EPR – a safe and competitive solution for future energy needs, IAEA-TECDOC-1117, December 1999.

Ganglaff, W. (1999) Westinghouse AP600 nuclear plant design, IAEA-TECDOC-1117, December 1999.

Haukeland, S., Ivung, B. & Petersen, T. (1999) BWR 90: the ABB advanced BWR design, IAEA-TECDOC-1117, December 1999.

Hopwood, J.M. (1999) The next generation CANDU 6, IAEA-TECDOC-1117, December 1999.

Huang, X. & Zhang, S. (1999) Advanced passive PWR AC-600: development orientation of nuclear power reactors in China for the next century, IAEA-TECDOC-1117, December 1999.

IAEA-TECDOC-968 (1997a) Utility requirements and safety objectives for advanced water cooled reactors, status of advanced light water cooled reactor designs 1996, September 1997.

IAEA-TECDOC-968 (1997b) Status of advanced light water cooled reactor designs 1996, September 1997.

IAEA-TECDOC-1117 (1999) Evolutionary water cooled reactors: strategic issues, technologies and economic viability, summary, December 1999.

Kim, I. & Kim, D.S. (1999) Major NSSS design features of the Korean next generation reactor, IAEA-TECDOC-1117, December 1999.

Matzie, R.A. & Ritterbusch, S.E. (1999) Evolutionary design and construction – the System 80 + ™ solution to the cost-benefit dilemma, IAEA-TECDOC-1117, December 1999.

Meyer, P.J. (1992) Design features and safety goals of the French–German PWR, IAEA Consultancy on Comparison of Advanced Light water Reactor Design Approaches, Vienna, IAEA-CT0764, 7–11 September 1992.

Nishimura, N., Nakai, H. & Ross M.A. (1999) The development of ABWR, IAEA-TECDOC-1117, December 1999.

Yadigaroglu, G., Berkovich, V., Bianchi, A., Chen, B., Meseth, J., Vecchiarelli, J. & Vidard, M. (1998) Advances in technologies for decay heat removal, IAEA-TECDOC-1117, Evolutionary Water Cooled Reactors: Strategic Issues, Technologies and Economic Viability, Seoul, December 1998.

Yamaguchi, H. Aeba, Y. & Weiss, E.H. (1999) Design features of APWR in Japan, IAEA-TECDOC-1117, December 1999.

Yu, S.K.W. (1999) Evolutionary CANDU 9 plant improvements, IAEA-TECDOC-1117, December 1999.

Chapter 11

Passive Systems and Inherent Safety

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Chapter 11
Passive Systems and Inherent Safety

11.1. INTRODUCTION/OBJECTIVES

The purpose of this chapter is to focus on how the design basis for evolutionary water reactors is being extended. The approach continues to be based on the defence-in-depth but a major difference is to attempt to include more severe (core melt) accidents within the design basis. This is achieved in evolutionary designs by the adoption of new technical features, not only to protect against present design basis events affecting the core and primary circuit, e.g. loss of cooling accidents (LOCA), steam line break (SLB) and steam generator tube rupture (SGTR) but also ultimately to protect against early and late containment failure.

Many evolutionary plant designs incorporate passive safety systems in place of active systems but in other respects do not vary substantially from current generation designs. In this chapter, the focus is again on water reactor technology for power generation since these reactors are such an important class of interest. Reviews of advanced light water reactor designs are given in IAEA-TECDOC-968 (1996), covering evolutionary medium- and large-size reactor designs for power generation. Further review of evolutionary designs including strategic issues and economic viability is given in IAEA-TEC-DOC-1117 (1999). A common feature is that decay heat is removed from the primary circuit to large tanks or pools via natural circulation. There are some new phenomena associated with decay heat removal in advanced designs with such components that are not found in present generation reactors. These are discussed in Relevant thermal–hydraulic aspects of advanced reactor design (1996). An issue here for the plant designer is to ensure that such systems have sufficient heat capacity and also initiate as intended. In addition, reactor coolant inventory is maintained using passive injection rather than active pump injection.

Different containment designs have been proposed, utilising steel, concrete or composites. Heat removal may need to be via natural circulation cooling of the containment wall in the case of steel or enhanced in concrete based containments using passive heat exchangers. These and other passive systems are covered in this chapter.

The design basis for the containment has traditionally been that it must survive the peak pressure arising from a double-ended guillotine break of the largest primary or secondary pipes. The design basis for more advanced plants will have to cover a broader selection of accident sequences, perhaps including significant core melting. This selection will be based on a combination of probabilistic and deterministic analyses. The lowest probability high consequence sequences will still need to be covered by engineering judgement or other means. There will be a tendency for deterministic analyses to be carried out by best estimate rather than conservative methodologies.

Advanced evolutionary water containments include other measures to ensure they survive under severe accident loads. Measures (IAEA-TECDOC-752, 1994) are introduced to prevent fuel coolant interactions (FCIs) to prevent direct containment heating (DCH) and to control hydrogen. They are also designed to reduce the source term by improving leak tightness. This is achieved via inherent safety features in the design, utilising passive heat removal systems in many cases. In addition to internal events, external events such as aircraft crashes and seismic events are also receiving special attention.

A number of more revolutionary designs of water reactor have been put forward as 'inherently safe' designs. These eliminate almost entirely active systems, e.g. relying on reactivity control via careful management of boron concentration. Some of these approaches are also summarised briefly although these are unlikely to be developed further at the present time.

11.2. ACTIVE HEAT REMOVAL SYSTEMS

The dissipation of decay heat is accomplished in present generation water reactors via redundant and diverse emergency core cooling systems (ECCS). One approach in evolutionary reactor development, both ALWRs and AHWRs, is to utilise the best features of these present systems in an optimal way (Yadigaroglu *et al.*, 1998), without significant recourse to new passive systems. Reactors based on this approach employ:

- improved system design with more redundancy, separation and diversity;
- increased pressure vessel water inventory;
- increased volume of pressuriser;
- direct in-vessel injection of cooling water;
- design features to reduce the risk of a LOCA, e.g. elimination of primary circuit piping;
- improved containment water storage tank facilities;
- introduction of cavity water flooding facilities
- automatic depressurisation of primary system followed by low pressure safety injection; and
- utilisation of a fire water system for containment sprays.

Plants in this class include EPR, ABWR, BWR 90, System 80+ and KNGR. There are in addition some CANDU and VVER designs. Table 11.1 summarises a few of the design highlights of these reactor types, which have been developed from optimisation of the best features of present generation plant.

Table 11.1. Classical evolutionary water reactor systems

Reactor	Description
EPR	Improved decay heat removal via active systems, e.g. ECCS
System 80+	
KNGR	Greater redundancy, diversity, independence, and separation of safety systems
ABWR	
BWR90	
VVER-1000	
CANDU 6, 9	Improved containment cooling systems

Yadigaroglu *et al.* (1998).

11.3. PASSIVE FEATURES

An alternative approach is to take more advantage of inherent forces such as gravity in the design of safety systems.

An additional advantage is that this approach results in simplified systems since it can eliminate the need for some redundancy, e.g. in emergency power supply systems. Some of the important reactor types in this category are listed in Table 11.2. Details of the various passive features of these plants were given in the previous chapter.

In evolutionary passive designs there are three typical components to protect against faults and accident conditions:

- cooling of the core via natural circulation, e.g. in intact circuit faults;
- gravity-driven cooling systems to mitigate LOCAs, and;
- passive containment cooling systems (PCCS).

In passive systems, the active components, e.g. pumps, diesels, fans, etc. are dispensed with and so there is no need for the redundant active safety grade systems associated

Table 11.2. Evolutionary water reactors incorporating passive systems

Reactor	Description
AP600, 1000	Innovative decay heat removal via passive systems
EPP	
SWR 1000	Utilisation of natural forces and phenomena, e.g. gravity, natural circulation, passive injection
ESBWR	
VVER-640	Passive containment cooling
CANDU 6(E)	

Yadigaroglu *et al.* (1998).

Table 11.3. Passive plant reduction in components

50% fewer valves
35% fewer pumps
80% less pipe (no safety grade pumps)
80% fewer heating, ventilating and cooling units (safety grade)
35% less seismic building volume
70% less cable

AP1000: Set to Compete (2002).

with active systems. This leads to simplification and hence potential scope for capital cost reduction. Similarly, the main ultimate heat sink is often the ambient air and hence there is no need for active service water systems.

Passive safety systems do not require the framework of safety support systems that are needed for current generation plant, including AC power, HVAC, cooling water systems and associated seismic buildings containing these components. Table 11.3 illustrates the consequential reductions in a number of key components compared with current generation plant (AP1000: Set to Compete, 2002).

Examples of evolutionary passive plants also include both light and heavy water reactors: PWRs – AP1000, AP600, EPP; BWRs – SWR 1000, ESBWR; and HWR – CANDU 6(E).

11.4. REACTOR COOLANT SYSTEMS

The purpose of the reactor coolant system is to maintain cooling during normal operation and also during transients. There must be sufficient water inventory and safety water injection systems to ensure that water reaches the core. Heat is then transferred by circulation (forced or natural circulation) to the ultimate heat sink (Fil *et al.*, 1999).

In many advanced plants, water to replenish any reduced water inventory in the primary circuit is stored entirely inside the containment. This provides additional protection against external events and other types of accident, e.g. containment bypass. Other features that are included to ensure protection of the primary circuit inventory include:

- pressurizer relief to a water storage tank;
- heat rejection to a water storage tank via heat exchangers;
- water storage tank joined with the containment sump;
- water storage tank, located high above the core for gravity driven injection, and;
- core make-up tanks (CMTs) at full circuit pressure to provide high pressure injection.

High-pressure passive injection systems are not present in currently operating reactors. The CMTs provide this function for AP1000 and AP600, (written AP1000/600 in this section). If the initiation set points are reached, valves open and cold water from the CMT flows into the reactor coolant system. If the CMT water level falls too low, then stepwise depressurisation of the reactor coolant system is initiated to ensure that medium and low pressure systems initiate.

Passive injection from accumulators is available in advanced passive designs, as it is in present generation plant. Modern designs have been optimised to increase system reliability and to broaden the pressure window of operation. Examples of such plants are AP1000/600, Mitsubishi APWR and Indian HWR designs. In addition, the Russian W-392 and W-407 designs adopt this principle. The Mitsubishi APWR accommodates an advanced accumulator system which eliminates the need for low pressure injection.

Passive low-pressure injection from the water storage tank is placed at high elevation across the core. Discharge can only take place when the reactor system pressure is at the last stage of depressurisation. Examples of such designs are AP1000/600 and the VVER-640/W-407 designs.

In advanced systems, sufficient heat transfer is attained provided there is sufficient water to cool the core. It is ensured by natural circulation from the heat source (core) to the heat sink, e.g. water storage tank in the AP1000/600 designs or the SGs in the Russian VVER-1000/W-392 design. These paths can exist in single- or two-phase water/steam modes. Different designs can make use of a range of different natural circulation paths.

Under accident conditions, heat is transferred to water tanks inside or outside the containment. Heat is then transferred to the surrounding atmosphere either via the containment shell or via a special heat exchanger, discussed below.

Passive feedwater systems have been considered in connection with the CANDU reactor design. There is an elevated tank above the boilers. Valves are opened to depressurise the boilers and allow flow by gravity.

11.4.1 Intact Circuit Decay Heat Removal

In an intact circuit accident, the heat sink is no longer available, e.g. to the steam generator secondary side or to the turbine.

In the AP1000/600 designs, under accident conditions, heat is transferred to the in-containment refuelling water storage tank (IRWST) via a passive residual heat removal heat exchanger (PRHR HX). This is connected to the reactor cooling system forming a full pressure, closed, natural circulation cooling loop (Hochreiter, 1992).

In, for example, a loss of normal feedwater scenario, the PRHR can remove sufficient heat to prevent operation of the pressurizer safety valves. The PRHR HX is activated following reactor trip and loss of power. If the pumps are operating, the flow through the passive RHR heat exchanger will be forced convection from the higher pressure cold leg to the hot leg. However, if the pumps are not operating, the flow direction will be reversed and by natural circulation from the hot leg to the top of the PRHR heat exchanger to the cold leg.

The EP 1000 incorporates a similar system (Yadigaroglu *et al.*, 1998).

Other designs are summarised in Yadigaroglu *et al.* (1998).

In the SWR 1000 design, there are Emergency Condensers connected to the RPV without valves and immersed in the core flooding pool.

In all the above cases, a further step is required to transfer heat from the pools to the ambient. These are described later in the containment section.

Other designs, e.g. KNGR Chang *et al.* (1997), some CANDU systems and some Siemens systems utilise cooling of the secondary side via a condenser.

The VVER-1000 and AC-600 systems make use of condensers outside the containment via a natural circulation air-cooled system.

Finally, the ESBWR and the Indian heavy water moderated light boiling water cooled AHWR utilise isolation condensers condensing steam from the RPV.

The passive cooling of the moderator in CANDU reactors employs a similar approach.

11.4.2 Passive Safety Injection

Various water sources are available inside the containment to provide decay heat removal and to protect the plant against LOCAs. These have been introduced earlier, e.g. the AP1000/600 and EPR have CMTs, high pressure accumulators, lower pressure CRTs and IRWSTs.

The systems need to provide protection against a spectrum of breaks. Since these systems are gravity driven or driven by overhead gas pressure, it is necessary that the reactor pressure be at a sufficiently low level for injection to take place. One way that this is achieved is by intentional automatic depressurisation via the reactor ADS. These systems are fitted in AP1000/600 and EPP reactors. Such systems were also fitted to present generation BWRs. Once the plant has been depressurised, low-pressure safety injection systems are then actuated. The AP1000/600 uses the IRWST inventory to provide reflood by gravity. Low-pressure gravity core make-up systems are available on ESBWR and SWR 1000.

The other approach is to increase the high-pressure coolant injection system capacity. The ABWR has an increased high-pressure coolant injection system.

With regard to water inventory, several of the new designs have a provision to flood the reactor to a level above the fuel. One such design is BWR 90.

The propensity of LOCAs is actually reduced in many new designs because of the elimination of certain components of primary system piping. For example, in the AP1000/600 and EPP, the recirculation pumps are situated in the steam generators. Similarly, internal pumps are used on the ABWR to eliminate recirculation piping.

11.5. CONTAINMENT

In the initial phase of an accident with heat released to the containment, credit can usually be taken from the heat sink associated with large structures inside the containment and from the containment itself. In the longer term, these structures will equilibrate with the containment atmosphere and new heat removal systems are required to take heat away to the ultimate heat sink.

There are various types of containment design (Hochreiter, 1992). Some designs are based on steel primary containments, e.g. AP1000/600 and the APWR but others rely on concrete primary containments, e.g. SBWR (re-enforced concrete and the EPR (pre-stressed concrete). Steel containments give the benefit of good heat removal characteristics and often incorporate passive heat removal concepts. Concrete containments have a proven capability to withstand greater loads but at the expense of poorer heat transfer characteristics. Concrete containments require additional heat transfer systems, e.g. heat exchangers or condenser systems to assist in heat removal from the interior to the exterior of the containment. This approach is adopted in EP 1000 (Cavicchia *et al.*, 1997).

The end objective in extending the design basis for the mitigation of severe accidents is to limit the radiological release to the atmosphere, i.e. to reduce the source term. The obvious way to achieve this is to maintain the structural integrity of the containment, to engineer isolation of penetrations and large passages and to prevent containment by-pass sequences. If the containment function remains intact then the radiological impact will be relatively minor, certainly to the general public.

Containment performance has been segregated into different categories:

- early containment failure; this might be caused by high-pressure vessel failure and DCH, in-vessel or ex-vessel steam explosions, local or global hydrogen deflagration or possibly detonation; failures to isolate or reactivity excursions;
- late containment failure; caused by melt attack on the containment structures or pressure boundary, or long-term pressure and/or temperature increase inside the containment;
- containment bypass; interfacing LOCA, SGTR.

Some of the measures and strategies under consideration in advanced designs are discussed below. There are essentially two approaches. Either the design can be improved to withstand the loads, the loads have to be reduced or possibly a combination of both solutions can be adopted. An example of the former might be to strengthen the containment, in the latter case, a high pressure melt ejection might be avoided by earlier system depressurisation.

Measures to meet the challenges to the containment are discussed below. The phenomena relate to pressure and temperature increase associated with decay heat and gas release from a molten core, high-pressure vessel failure and DCH, steam explosions, hydrogen detonation, melt attack on the vessel pressure boundary and containment structures, and reactivity accidents (Ward, 1992).

11.5.1 Measures to Control Pressure and Temperature

There are various measures that are included in advanced containment design for the control of pressure and temperature. The objective is to limit the pressure to below an acceptable limit and to reduce the pressure down to atmospheric pressure as quickly as possible to limit fission product release to the environment. The systems should be passive so that they can still function reliably under severe accident conditions. The heat loads can arise from decay heat and in the event of a severe core meltdown from the Zircaloy/steam exothermic reaction and also possibly from MCCIs.

11.5.1.1 Passive Containment Cooling. This method is used with a steel containment with good heat transfer characteristics. Pressure reduction using this system will be relatively slow and will also depend on the partial pressure of non-condensable gases in the containment. External cooling is enhanced by external sprays.

The AP1000/600 containments comprise an inner steel containment shell surrounded by an exterior concrete shield building, Figure 11.1. The inner steel containment not only acts as a barrier to radioactive release but also serves as an integral part of the heat release system. It is prevented from over heating by a PCCS that provides a natural circulation draught of air cooling between the steel containment shell and the shield building (Scobel and Conway, 1990). This serves to enhance the heat removal from the PCCS.

A similar approach is adopted by the simplified PWR (SPWR) of Westinghouse–Mitsubishi design. This design has been scaled up to 900 and 1200 MW units (Lillington and Kimber, 1997; Naitoh *et al.*, 1992).

Other conceptual designs (Lillington and Kimber, 1997; Kuczera, 1992) for example have been put forward by KfK, which consist of an inner steel containment surrounded by a strong re-enforced concrete wall. Both the inner steel and the concrete walls share the loads. There is an annulus between the two shells through which air flows by natural circulation.

11.5.1.2 Condenser Systems. In this system, heat is transferred to the external atmosphere via an intermediate circuit, which carries single- or two-phase water under natural or active system circulation. As for the PCCS described above, the effectiveness of heat transfer will depend on the partial pressure of non-condensables inside the containment. Pressure reduction is also relatively slow.

In the EP1000, a finned condenser is located at the top of the concrete primary containment. This transfers heat via a thermosyphon loop through the concrete containment walls to an external heat exchanger located in a tank. This is initially immersed in water but later in the accident is air-cooled (Yadigaroglu *et al.*, 1998).

The cooling of the containment atmosphere by a condenser is also proposed for the SWR 1000 design. This transfers the heat to a secondary system connected to an external pool.

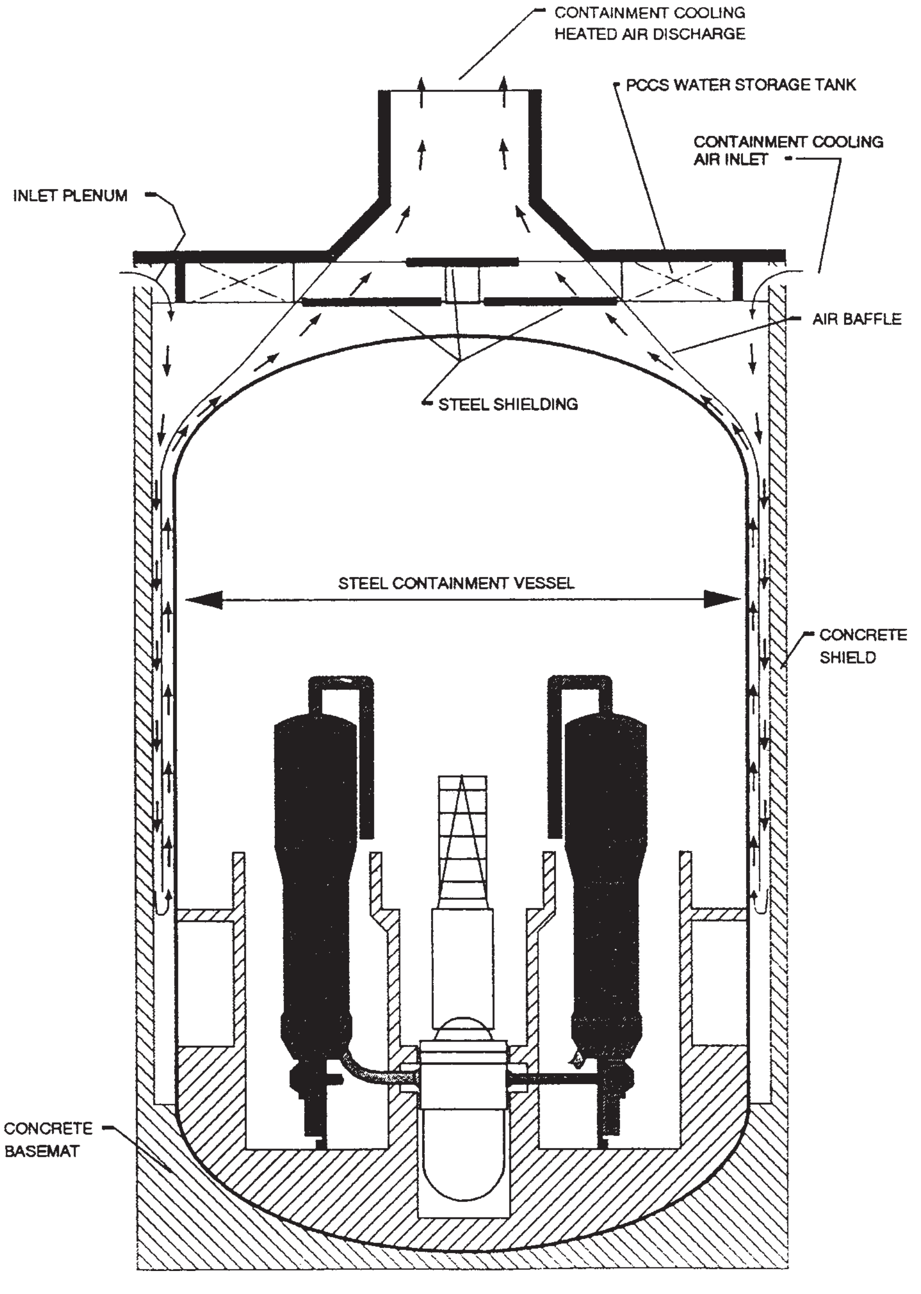

Figure 11.1. AP600 Passive containment cooling. Source: Scobel and Conway (1990) and Ambrosini (1992).

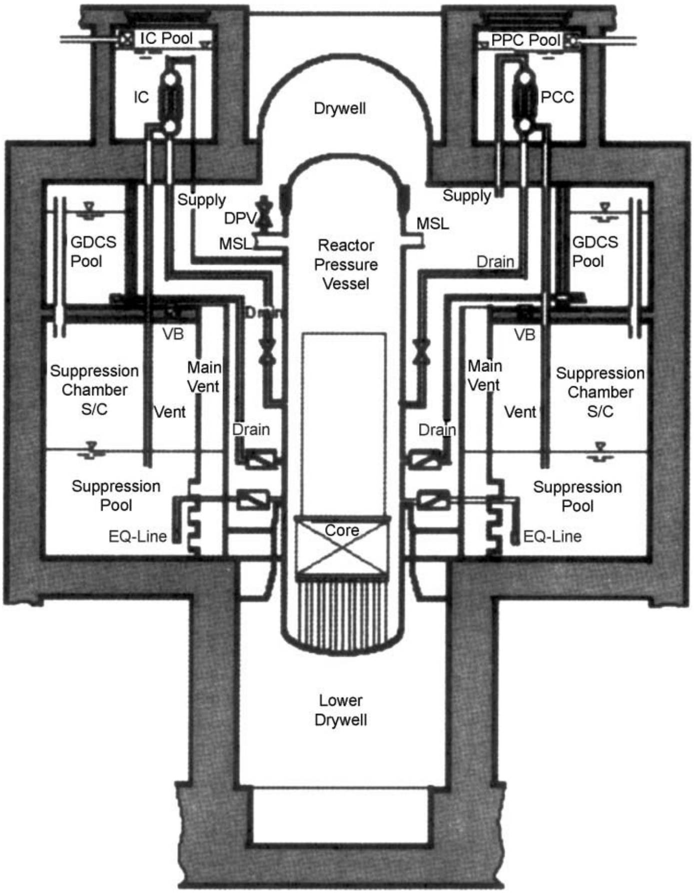

Figure 11.2. ESBWR Passive core and containment cooling. Source: Yadigaroglu *et al.* (1998).

In the CANDU 6 system, a containment condenser transfers heat to a secondary side connected to the Passive Emergency Water System Tank (Hopwood, 1999).

In the ESBWR (Orsini and Pino, 1992), a PCCS is incorporated into the design of the containment to remove decay heat from the drywell (Figure 11.2). In this system, containment steam is condensed in an external pool. Non-condensables are discharged to the suppression pool.

11.5.1.3 Internal Containment Sprays. Internal spray systems may have both significant advantages but also disadvantages. The present designs tend to have active components and are, therefore, susceptible to not functioning in a hostile environment. Passive systems have been considered but have reduced capacity and may not function correctly in the presence of aerosols. Sprays will also not reduce the pressure if there is a significant partial pressure of non-condensables, e.g. hydrogen from metal water reactions or other gases from core concrete interactions.

11.5.1.4 Sump Water Cooling. Systems to reduce the containment pressure by cooling the sump water are another possible method. However, there needs to be good natural circulation cooling of the sump water which is necessary for effective heat removal.

11.5.2 Measures to Control FCIs

The issue of in-vessel FCIs has been postulated in the context of present generation reactors. The defence strategies are an attempt to exclude this possibility by design or to demonstrate that the vessel will not fail or demonstrate that the containment remains intact after vessel failure. In advanced reactors, if failure of the vessel is assumed, there is the opportunity to design a reactor cavity that can survive the load (EIBL *et al.*, 1992), and also to protect the containment from flying missiles by including an upper shield or slab.

It is generally expected that there may be a greater possibility of vessel failure if the system has been depressurised. Depressurisation is often a strategy in plants with passive injection to insure that injection can occur and so in principle in-vessel FCIs may be an issue for some advanced plant designs. However, some analysts believe that steam explosions in-vessel will not be sufficiently energetic to cause vessel failure.

The possibility of ex-vessel FCIs can be substantially reduced by preventing molten core material exuding from the bottom of the vessel from coming into contact with water. A number of preventative features are proposed in current advanced designs.

The ‘core-catcher’ has been proposed for the EPR, for example, Figure 11.3. In this design, the melt is spread horizontally over a large dry area of about 150 m^2. Once in this

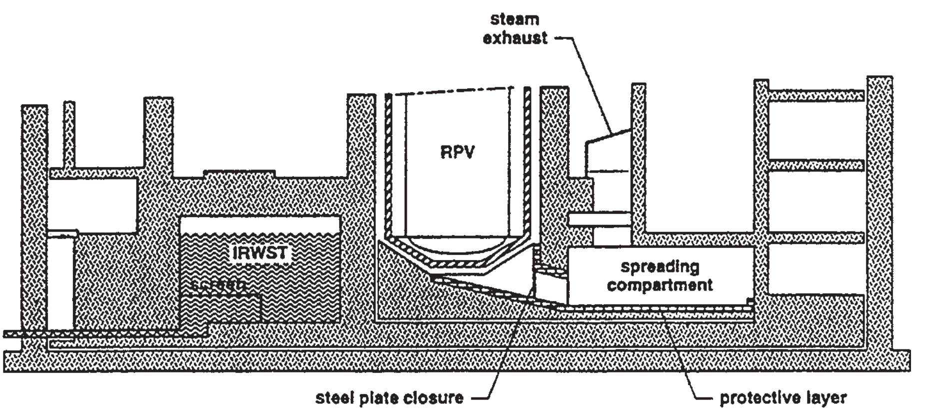

Figure 11.3. European pressurised reactor. Source: Leverenz (1999).

spreading compartment, the corium would then melt through various low melting point plugs that would eventually let water through from a large IRWST tank to flood the corium (Leverenz, 1999). Heat would be dissipated from the melt by evaporation for a 0.5–1 day period, after which an alternative containment cooling system would come into operation. In the case of EPR, this involves containment sprays, cooling of the water in the spreading compartment and also cooling of the IRWST water.

Other types of core catcher have been proposed. These include a similar ‘plug melt-through’ concept into crucibles in a dry vertical core catcher concept. The crucibles are then cooled by natural circulation of water, which is ultimately discharged through the containment to an ultimate heat sink. Another type radiates heat to a large conducting surface in the reactor cavity, which is then cooled by external natural circulation.

The retention of core melt has been investigated in several experimental programmes. This includes programmes in Germany and the MACE tests in the US.

In another German design (Kuczera, 1992), the corium is allowed to fall into a dry cavity with a thin bottom layer of low melting point material. Hollow plugs are eventually uncovered allowing water to flow up the plug holes and cover the corium.

Another variation of design to achieve cooling is to have a staggered pan arrangement in an oxidic ceramic bed. The upper part of this bed remains dry and the lower part is flooded with water. Heat is extracted via natural circulation of water through the particle bed. The possibility of steam explosions is reduced because the top part of the bed remains dry.

The other way to ensure that melt does not come into contact with the water is to prevent the vessel failing. One postulated approach is to flood the vessel in the reactor cavity. The effectiveness of this measure will depend on the power density and the geometry of the vessel (surface area). In this method, heat is removed from the melt via conduction through the lower head of the vessel.

11.5.3 Control of Hydrogen

The control of hydrogen and the possible back-fitting of safety systems are live issues in the operation of present day reactors. The sources include the Zircaloy/steam (and some contribution of steel/steam) reactions and also contributions from metal reactions in MCCIs. There may also be a contribution to the hydrogen source from long-term radiolysis of water.

The design measures that can be taken include the presence of a large containment volume. This helps to keep hydrogen concentration levels to below detonation limits (10–13%). There also needs to be good mixing preferably by natural circulation. There should be a minimal number of compartments to prevent the build-up of local hydrogen concentration.

In some designs, the containment atmosphere can be pre-inerted. This may be a useful approach for smaller containments but a disadvantage is that the containment is inaccessible during operation. For such containments, special attention will need to be paid to gas build-up by radiolysis during the periods when the containment is not de-inerted, e.g. during refuelling.

Various hydrogen reduction devices are available. Igniters, catalytic or battery-driven recombiners are useful for dissipating the hydrogen at lower concentrations but are less suitable for large concentrations. Recombiners may also be appropriate for longer term hydrogen management.

11.5.4 Reduction of Source Term

Advanced plants have a number of design features, active systems and attributes relying on natural processes to reduce the source term in the event of vessel failure.

Clearly corium emanating from the vessel should be appropriately quenched (in such a way to avoid a steam explosion). Ways in which this can be achieved have been discussed above.

MCCIs result in the emission of a large quantity of aerosols that carry fission products into the containment atmosphere. Possible measures to reduce MCCIs, e.g. using 'core catchers' have also been discussed above.

Large surface areas are useful for the plate out of aerosols. There are many natural processes, agglomeration, sedimentation, diffusiophoresis, thermophoresis and hygroscopicity that promote deposition onto surfaces.

Internal containment sprays provide a means of entraining or dissolving air-borne fission products in water which can then be retained in the containment sump. There are chemicals such as sodium hydroxide, sodium thiosulphate or hydroxine that can be put into the water in the spray systems to enhance the removal of some fission products, especially iodine and caesium.

Fission products can be scrubbed in large pools of water. Similarly, water flooding of debris also provides a potential for scrubbing.

Elemental iodine resuspension can be reduced by the maintenance of a pH > 7 in water pools.

In the SBWR design (Naitoh *et al.*, 1992), steam released to the drywell is channelled through a condenser. It is then condensed and then returned to the gravity-driven cooling system pools. This provides a mechanism for aerosol deposition and fission product removal.

The source term can be mitigated in some designs by introducing ventilation systems for cleaning exhaust air. The SPWR, which is a variant of the AP600, developed by Westinghouse and Mitsubishi, includes in its design an emergency passive air filtration system to mitigate releases into the lower containment penetration area. The air is filtered before being mixed with the cooling air of a PCCS system (similar to that in AP1000/600).

Ventilation systems may also be useful for designs with a secondary confinement if it became contaminated as a result of leakage from the primary containment. In some cases, primary containments are surrounded by additional containment buildings maintained at a slightly sub-atmospheric pressure. This is to ensure that residual fission products released from the primary containment do not escape. Controlled release from filters or stacks may then be considered.

11.5.5 Leakage Control

Advanced reactors can be designed to include a number of features to improve leak-tightness under severe accident conditions. Under such conditions, the containment structures will have to withstand much higher loads in terms of pressure, temperature, radiation and chemical attack than under normal operation. Leak-tightness must be assured under this harsher environment; leaks may occur from the containment structures themselves, pipe and electrical penetrations and isolation valves, hatches, locks, etc.

Leak rates from steel containments or containments with a steel liner are expected to be lower than those from concrete containments without a liner. However, in some designs, the concrete containment is surrounded by a secondary containment. There is generally a requirement to improve overall leak-tightness in advanced containment designs, which can be achieved by improved primary containment design or possibly by taking credit for a secondary containment without improvement to the leak-tightness of the primary containment. By way of example of improved leak-tightness, the design leak rate for AP600 is 0.12% per day against other current PWR rates which are in the range 0.25–0.5% per day (IAEA-TECDOC-752, 1994). Containment leak-tightness needs to be maintained under all plant states including shutdown.

There are containment bypass sequences such as interfacing LOCAs and SGTRs and leaks in these events also need to be covered. The general approach in advanced containment design is to reduce the number of penetrations. Other special measures include pressurisation systems that keep penetrations at pressures higher than the containment pressure. These systems have been proposed in plants where there is a direct release path to the environment. Suction systems have been proposed to collect the leak contents and treat it before release. Other special components such as bellow's valves and seal welds on large equipment hatches are also considered. With all these systems, however there are issues concerning their likely performance under severe accident conditions.

There are issues such as how the leak-tightness of the containment under severe accident loads can be tested. This is particularly so if the severe accident postulated pressures are higher than the peak design pressure, so periodic testing is not possible.

Systems have also been proposed for establishing whether a large opening in the containment boundary is pre-existing when an accident occurs. This approach has been considered for existing plants. One such system, developed by EDF gives a measure of leak-tightness by measuring the rate of increase on containment pressure cause by the usual leaks in the air supply system.

11.6. REVOLUTIONARY DESIGN CONCEPTS

There are designs proposed that are radically different from current generation technology and these would require substantial development and investment before building and licensing. Examples are given in Table 11.4.

Integral type pressurised water reactors such as PIUS, VPBER-600, SPWR and ISIS (PIUS, 1997; VPBER-600, 1997; SPWR, 1997; ISIS, 1997) are completely immersed in a large pool. Many of these concepts have been described as inherently safe, i.e. they depend entirely on the forces of gravity and natural circulation for operation. Typical design objectives are that they should be ‘operator forgiving’ and should incorporate simple safety principles (which should therefore imply increased reliability). For flexibility of supply and operation they should be available in small- or moderate-size units, which could be coupled if necessary. Such designs and other revolutionary approaches are considered in this chapter.

The PIUS reactor (PIUS, 1997) is immersed in a large pool where the boron concentration is controlled by several ‘density lock’ arrangements (Figure 11.4). There are no control rods and the required reactivity is maintained by control of the boron concentration and moderator temperature. In the event of an accident, a natural circulation loop through the core is established, resulting in reactor shutdown and core cooling.

The VPBER-600 (VPBER-600, 1997) is an integral PWR, located in a guard vessel. The design basis was taken from the AST-500 heating reactor, which was designed in the early 1980s. VPBER-600 includes passive safety systems and diverse operation principles with significant redundancy and self-actuation.

Table 11.4. Advanced revolutionary reactors

Reactor	Design organisation	Capacity (MWe)
PWR		
PIUS	ABB, Atom	650
VPBER-600	OKBM	630
SPWR	JAERI	600
ISIS	Ansaldo Spa.	300
JPSR	JAERI	630

Data from IAEA-TECDOC-968 (1997), PIUS (1997) VPBER-600 (1997) SPWR (1997), ISIS (1997) and JPSR (1997).

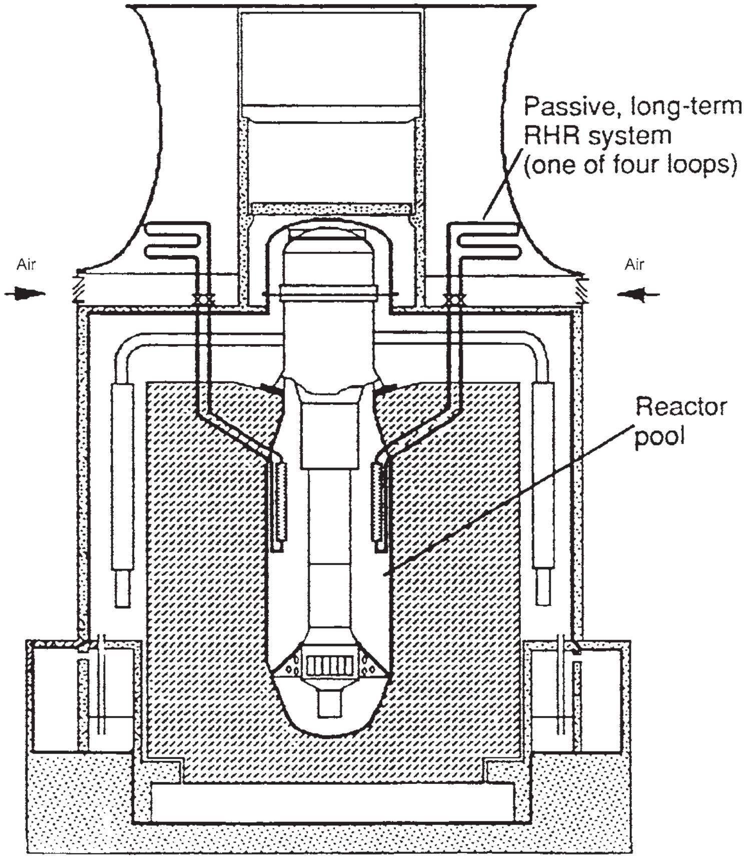

Figure 11.4. PIUS. Source: PIUS (1997).

The SPWR (SPWR, 1997) is based on an integral design with the complete primary circuit including the core, MCPs, pressuriser and the SG encompassed within the reactor pressure vessel. It employs passive systems for shutdown and decay heat removal under normal operation and also accident mitigation. Highly borated water is used for shutdown in place of control rods.

ISIS (ISIS, 1997) is also an integral PWR reactor, which is completely immersed in cold borated water. It is similar to the PIUS concept except that the reactor components are derived from proven technology.

Unlike the integral designs described above, the JPSR is a passive two-loop PWR design (JPSR, 1997), adopting a boron-free concept to increase reactivity sensitivity to changes in moderator density. As a result, reactor power can be controlled by adjusting the steam generator feedwater flow rate. This simplification in design results in a reduction in manpower for operation and maintenance.

REFERENCES

Ambrosini, W., Manfredini, A., Oriolo, F. & Paci, S. (1992) The influence of containment response on primary system accident evolution in innovative light water reactors, IAEA-TC-818, Aix-en-Provence.

AP1000: Set to Compete (2002), Nuclear Engineering International, October 2002.

Cavicchia, V., Fiorino, E. & Vanini, P. (1997) Innovative containment cooling for a double concrete containment, *Proceedings of the International Meeting on Advanced Reactors' Safety – ARS-97, (Orlando, FL, 1–5 June 1997), American Nuclear Society*, pp. 1305–1312.

Chang, S., No, H.C., Baek, W.P. & Lee, S.I. (1997) Korea looks beyond the next generation. *Nucl. Engng. Int.*, 12–16.

Fil, N.S., Allen, P.J., Kirmse, R.E., Kurihara, M., Oh, S.J. & Sinha, R.K. (1999) Balancing passive and active systems for evolutionary water cooled reactors, IAEA-TECDOC-1117, Evolutionary water cooled reactors: strategic issues, technologies and economic viability, December 1999.

Hochreiter, L. (1992) Thermal–hydraulic aspects of AP600 passive safety systems design, NURETH5, Salt Lake City, US, September 1992.

Hopwood, J.M. (1999) The next generation CANDU 6, IAEA-TECDOC-1117, Evolutionary water cooled reactors: strategic issues, technologies and economic viability, December 1999.

IAEA-TECDOC-752 (1994) Status of advanced containment systems for next generation water reactors.

IAEA-TECDOC-968 (1997) Status of advanced light water cooled reactor designs 1996, September 1997.

IAEA-TECDOC-1117 (1999) Evolutionary water cooled reactors: strategic issues, technologies and economic viability, December 1999.

ISIS (1997) Ansaldo, Italy, IAEA-TECDOC-968, Status of advanced light water cooled reactor designs 1996, September 1997.

JPSR (1997) JAERI/IHI, Japan, IAEA-TECDOC-968, Status of advanced light water cooled reactor designs 1996, September 1997.

Krieg, R., Alsmeyer, H., Eibl, J. *et al.* (1992) Extreme loading of inner structures of next generation PWR containments, *Proceedings of the Fifth Workshop on Containment Integrity*, NUREG/CP-0120, SAND92-0173, Washington, DC, 12–14 May 1992.

Kuczera, B. (1992) R&D investigations relating to future advanced PWR containment concepts, IAEA-TC-818.

Leverenz, R. (1992) Design description of the European pressurised water reactor, IAEA-TECDOC-1117, Evolutionary water cooled reactors: strategic issues, technologies and economic viability, December 1999.

Lillington, J.N. & Kimber, G.R. (1997) Passive decay heat removal in advanced nuclear reactors. *J. Hydraulic Res.*, **35** (6).

Naitoh, T., et al. (1992) Improved emergency elevated air release for simplified PWR. *Proceedings of the Tokyo International Conference on Design and Safety of Advanced Nuclear Power Plants, Atomic Energy Society of Japan, October 25–29, 1992*, vol. IV, Atomic Energy Society of Japan, pp. 25–29.

Orsini, G. & Pino, G. (1992) SBWR reinforced concrete structural performance during severe accidents, Proceedings of the Fifth Workshop on Containment Integrity, NUREG/CP-0120, SAND92-0173, Washington, DC, 12–14 May 1992.

PIUS (1997) ABB Atom, Sweden, IAEA-TECDOC-968, Status of advanced light water cooled reactor designs 1996, September 1997.

Relevant thermal–hydraulic aspects of advanced reactor design, OECD/GD, R (96) 22, November 1996.

Scobel, J.H. & Conway L.E. (1990) Performance of the Westinghouse AP600 passive containment cooling system under severe accident conditions, Second International Conference On Containment Design and Operation, Toronto, 14–17 October 1990.

SPWR (1997) JAERI/IHI, Japan, IAEA-TECDOC-968, Status of advanced light water cooled reactor designs 1996, September 1997.

VPBER-600, OKBM (1997), Russian Federation, IAEA-TECDOC-968, Status of advanced light water cooled reactor designs 1996, September 1997.

Ward D.A. (1992) Proposed criteria to accommodate severe accidents in containment design, *Proceedings of the Fifth Workshop on Containment Integrity*, NUREG/CP-0120, SAND92-0173, Washington, DC, 12–14 May 1992.

Yadigaroglu, G., Berkovich, V., Bianchi, A., Chen, B., Meseth, J., Vecchiarelli, J. & Vidard, M. Advances in technologies for decay heat removal, IAEA-TECDOC-1117, Evolutionary water cooled reactors: strategic issues, technologies and economic viability, Seoul, December 1998.

Chapter 12
Future Generation Reactors

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Chapter 12
Future Generation Reactors

12.1. INTRODUCTION/OBJECTIVES

This chapter considers the innovative reactor designs that are being put forward as likely candidates for future generations of reactors. Higher efficiencies can be achieved for electricity generation by increasing the temperature of the reactor systems. Higher temperatures are a feature of many of the most promising future reactor designs.

There is also the potential of exploiting nuclear energy for more general energy applications than have been considered previously. These applications could be many and varied. They include the utilisation of fuel cycle systems to burn weapons grade plutonium or minor actinides from spent fuel. The possibility of utilising nuclear energy to generate hydrogen is an attractive option for transport. Efficient future reactors for electricity generation and for these additional applications are considered in this chapter.

Similar design requirements to those described for evolutionary plants in Chapter 7, relating to reliability, economics, safety and acceptability apply also to these type of systems, together with some additional requirements. General design requirements for these future reactors are described in this chapter.

It is worth noting at this stage that sub-critical reactors based on accelerator driven systems (ADS) are also attractive candidates for plutonium destruction and minor actinide conversion. These are considered separately in Chapter 13.

Some of the innovative reactor designs reviewed in this chapter are also being considered for heat applications. Heat and other novel applications for nuclear energy are considered in more detail in Chapter 14. In Chapter 12, the focus is on the innovative reactor designs, in Chapter 14 the focus is on novelties in the applications. Already in some countries, e.g. Russia, waste heat from electricity generators is being used for district heating. Most of the experience to date has been with low-temperature applications. Other low-temperature applications include desalination plants. In many cases, the proposed reactor designs and certainly already operating systems are based on established reactor designs; the novel aspects relate to the balance of plant configurations to achieve the desired goals.

Many of the most promising future reactor designs have been examined by the US instigated Generation IV Forum (GIF) programme that started a little over 2 years ago. There are a number of signatories from among the major nuclear plant operating countries, 10 countries have joined, Argentina, Brazil, Canada, France, Japan, South Africa, South Korea, Switzerland, UK and the US. Other European countries are participating through the EU, which is also a member.

Table 12.1. Generation IV systems

System	Spectrum	Fuel cycle	Application
Supercritical water reactor (SCWR)	Thermal and fast	Once-through/closed	Electricity/actinide management
Very high temperature reactor (VHTR)	Thermal	Once-through	Electricity/hydrogen production/process heat
Gas-cooled fast reactor (GCFR)	Fast	Closed	Electricity/actinide management/ hydrogen/process heat
Sodium-cooled fast reactor (SFR)	Fast	Closed	Electricity/actinide management
Lead/lead–bismuth cooled fast reactor (LFR)	Fast	Closed	Electricity/actinide management/hydrogen
Molten salt reactor (MSR)	Thermal	Closed	Electricity/actinide management

IEA/OECD (NEA)/IAEA (2002) and The US Generation IV Implementation Strategy (2003).

The objective of Generation IV is to identify the most promising types of reactor design that will contribute to future generations of reactors and to put in place R&D to promote further understanding of the designs and their performance.

Initially over 100 different designs were considered under the simple title of future energy systems (not just nuclear). These were reduced to 19 designs and finally to the following 6 most promising designs, see Table 12.1.

There has also been a 'Three Agency Study' carried out by the International Energy Agency (IEA), the OECD Nuclear Energy Agency (NEA) and the International Atomic Energy Agency (IAEA) (IEA/OECD (NEA)/IAEA, 2002). There were 34 innovative designs considered. Of these, a total of 12 designs have been considered in some detail. Most of these are also included in the Table 12.1 categorisations.

12.2. SUPERCRITICAL WATER REACTORS

Ways in which to substantially enhance the efficiency of LWRs have been studied for some time. Efficiencies as high as 44% are possible by operating in a thermodynamically supercritical regime. Supercritical high performance reactors are one of the candidates of the Generation IV initiative (The US Generation IV Implementation Strategy, 2003) for medium term deployment. The European Commission is also currently assessing the merits and feasibility of such an approach in a project involving European institutes and industry in collaboration with the University of Tokyo (Squarer *et al.*, 2001). A review of supercritical reactors has been carried out by Oka (Proceedings of the First International Symposium on Supercritical Water-Cooled Reactors, 2000) and the EC project is assessing the available technology against a reference design (Dobashi *et al.*, 1998). There have been considerable advances in this technology in Japan.

Table 12.2. Supercritical water reactors

Reactor	Rating (MWe)	Country
Light water		
SCWR (Gen IV)	1700	GIF Members
SCLWR	1000	Japan
B-500 SKDI	515	Russia
Heavy water		
CANDU SCWR (Gen IV)	~1000	Canada
CANDU X	350–1150	Canada

Data from IEA/OECD (NEA)/IAEA (2002), The US Generation IV Implementation Strategy (2003), Squarer *et al.* (2001) and Silin *et al.* (1993).

Supercritical water reactor (SCWR) systems are principally aimed at electricity production. Their high thermal efficiency offers a potential for improved economics compared with current generation LWRs. An important issue in regard to these systems is the need to develop materials and structures that can serve in the high temperature and supercritical pressure regimes of these plants. A sample of designs currently under consideration is given in Table 12.2.

The concept is based on a once-through cycle, operating in excess of the water critical pressure of 22.1 MPa. Water enters the reactor core and then exits without change of phase. This system has the advantage that no steam–water separation is necessary, which in principle leads to a simplified (and therefore more economic design). Heat is removed from the system via a coolant of very high temperature and because the system is single phase, the turbines are driven directly by the primary coolant. Typically, water enters the core at about 280°C and exits at 500°C or higher, yielding efficiencies of the order of about 44%.

Supercritical systems have been considered at various times over the past 50 years, initially by Westinghouse and GE and in the last decade by Kurchatov Institute and AECL, based on a CANDU system. The early Westinghouse and GE designs were light water cooled. The Kurchatov and AECL designs were graphite moderated and heavy water cooled respectively; however, these required larger reactor volume and complicated systems. This resulted in less favourable economics. The Russian design, based on an integrated supercritical PWR design, was cooled via natural circulation but was more limited in scale and power. Heavy water super critical systems are considered below.

There have been various other types of supercritical reactor designs considered, including fossil plant systems, the GE nuclear super-heater, a steam cooled FBR (FZK), a B&W design, and a University of Tokyo steam-cooled FBR.

12.2.1 Light Water

12.2.1.1 SCWR (Gen IV). The Generation IV supercritical water cooled system is a thermal reactor aimed at electricity production as the primary option (Figure 12.1).

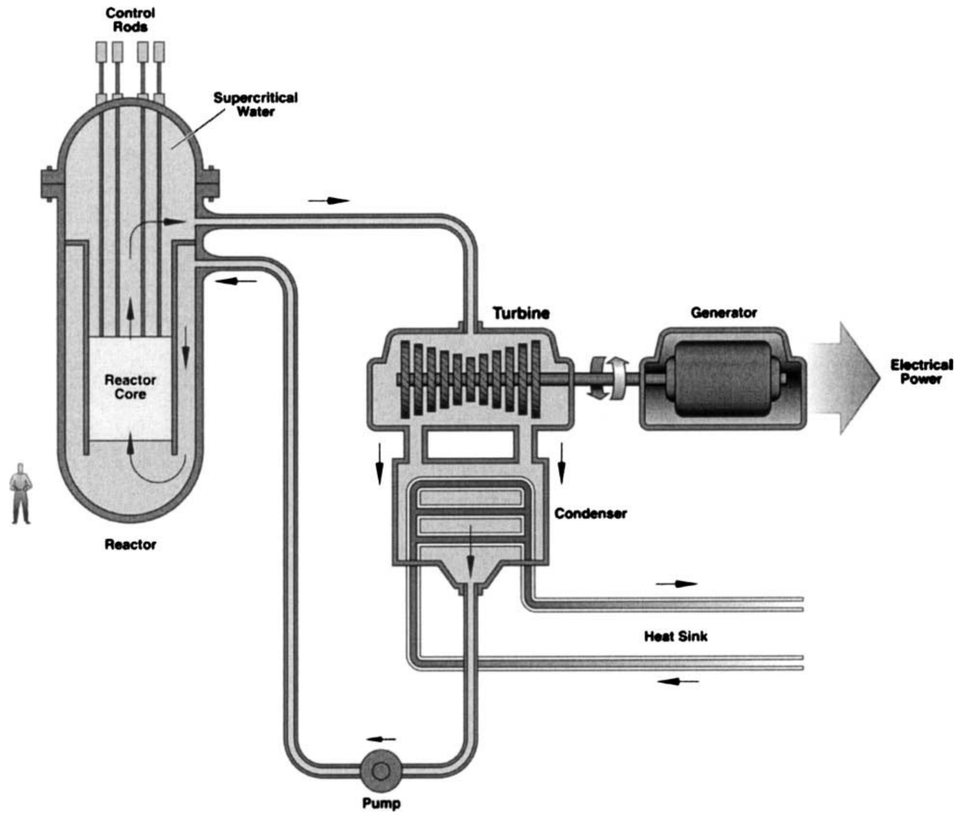

Figure 12.1. Supercritical water reactor. Source: NEA Annual Report (2002).

The option is, however, retained of converting the core design to a fast spectrum to enable actinide recycle.

The reference plant has 1700 MWe power at an operating pressure of 25 MPa above the thermodynamic critical pressure of water. The outlet temperature has a reference level of 510°C but this could range up to 550°C. The system has a high efficiency of 44% (The US Generation IV Implementation Strategy, 2003). This results in good economics, further enhanced via a simplified plant design. However, due to its corrosive high-temperature water environment, the SCWR requires significant materials development. Further developments are also required to address a number of operational safety issues.

12.2.1.2 SCLWR. The study (Squarer *et al.*, 2001) takes the University of Tokyo's Super Critical Light Water Reactor (SCLWR) as one of the most likely economically competitive of the proposed designs. It can also be designed as a fast reactor that could be fuelled with MOX fuel of around 12% enrichment.

The high-performance LWR size scales considered by Japan (Squarer *et al.*, 2001) are based on the following parameters for core and fuel design, reactor pressure vessel, containment, turbine and balance of plant. The scale is that of a large 1000 MWe power output, at 25 MPa pressure and 500°C outlet temperature. It has 4.2 m active core height, 121 fuel assemblies with 8 mm OD fuel pins, and control rods inserted from the top, 3.380 m RPV ID and 27.5 MPa design pressure, a cylindrical containment with a turbine frequency of 50 cycles s^{-1}. It can be seen that this design is consistent with the Generation IV reference design.

12.2.1.3 B-500 SKDI. The B-500 SKDI design from Russia (Silin *et al.*, 1993) is based on a natural circulation, integrated supercritical pressurised water reactor system, at a smaller scale and power; reference power is 515 MWe.

12.2.2 Heavy Water

12.2.2.1 CANDU SCWR (Gen IV). The SCWR concept is also being considered as an evolution from CANDU reactor technology. As with the LWR systems, there is an aim to continually enhance the design and applications of the CANDU system. Thus, complementary to the SCWR loop concept described above, a channel design option with multi-stream products is also possible within the SCWR context. The CANDU SCWR concept is envisaged for flexibility of application, e.g. including electricity production, hydrogen generation (direct or indirect) and high-temperature process heat applications, depending on demand. It could also have desalination applications (Generation IV Seminar on Nuclear Energy Systems Research and Development, 2004; Duffey, 2004a,b). It is seen as part of the evolution towards the CANDU X system, sometimes referred to as a Generation V system. The CANDU X design could also be economically competitive. The main elements of the CANDU X system are described below.

12.2.2.2 CANDU X. The CANDU X concept is another pressure tube reactor in the CANDU family of reactors. The design is being put forward by AECL in Canada. It has a flexible generating capacity, in the range 350–1150 MWe. This depends on the number of fuel channels in the plant.

The innovative features of the Mark 1 model include supercritical heavy water for the reactor coolant and supercritical light water for the turbine generator. The utilisation of supercritical water results in a significant increase in system pressure and temperature compared with earlier generation CANDU plants.

CANDU X retains the use of two passive shutdown systems as in current generation plants. There is also passive decay heat removal even if the reactor system is empty of coolant.

The CANDU X reactor possesses a number of the attributes expected from future generation systems. It has high efficiency due to increased core outlet temperature. There is flexibility in reactor power scale available through extensive modularity in design.

Regarding its fuel cycle and waste management concerns, the option to use thorium fertile material and slightly enriched fuels is available to reduce the level of minor actinides produced.

As for supercritical light water systems, the main applications would be for electricity generation. However, the high core outlet temperature increases the number of process heat applications that are possible. The temperature is higher than can be achieved by current water reactors but lower than can be achieved for HTRs and LMFRs. The attractiveness for process heat applications is particularly true for the smallest 350 MWe version.

12.3. HIGH-TEMPERATURE GAS-COOLED REACTORS

Gas-cooled reactors have been studied in various countries since the start of the nuclear power programme (Methnani, 2003; Mitchell *et al.*, 2002). Future generation plants will benefit from this experience. In this section, attention is focussed on the high temperature thermal systems, in the following section, fast spectrum systems will be considered. The early gas reactors were natural uranium fuelled, graphite moderated and air cooled and used for military operations. Following on, in the UK, Magnox plants incorporated pressurised carbon dioxide cooling followed by advanced gas reactors with enriched uranium oxide fuel and higher pressure carbon dioxide as coolant.

High temperature gas-cooled reactor (HTGR) concepts have been studied in parallel with the carbon dioxide-cooled plants. Early experimental and prototype reactors included Dragon, AVR and Peach Bottom. The Dragon reactor operated at Winfrith and incorporated helium cooling and ceramic-coated particle fuel. This reactor included highly enriched uranium–thorium carbide fuel particles. The coolant operating outlet temperature was 750°C and much useful information on helium-based HTGR systems arose from the early Dragon programme. The AVR system operated in Julich in Germany. It had a higher temperature of 950°C and used 100,000 coated fuel spheres. This was the concept that is currently being considered for the Pebble Bed Modular Reactor (PBMR) design. In this design, the fuel spheres move downwards in the reactor core within a graphite reflector vessel. The first HTGR in the US was Peach Bottom Unit 1, rated at 40 MWe. Several fuel designs have been developed to overcome problems with cracked fuel.

Two main types of HTGR designs have emerged over the last 2 decades, through the operation of several prototypes. The German thorium high-temperature reactor (THTR-300) was of a pebble bed type; The US Fort St. Vrain design was of the prismatic design.

Power ratings were raised to 300 MWe and there were various design features including a pre-stressed concrete reactor pressure vessel and a more advanced coated fuel particle design known as TRISO.

More recent designs have incorporated reduced power density, reduced overall power and more passive systems. The general atomics modular high temperature gas reactor (MHTGR) was rated at 350–450 MWt and the German HTR series design was rated at 200–300 MWt. These system designs were more modular. The direct cycle MHTGR design, utilising advanced gas turbine and high temperature turbine technology, could yield efficiencies up to 50%.

The IAEA has co-ordinated several safety-related research projects on the physics, heat removal aspects and fuel and fission product behaviour of HTGRs. A latest activity is concerned with benchmarking core physics and thermal–hydraulic methods against experimental data in order to evaluate HTGR performance.

The European Commission has recently supported a network R&D activity to address the major design issues associated with the core physics and fuel cycle, and the material and components issues. The project is also concerned with the safety and licensing issues associated with the HTGR design.

In respect of their reactor physics, HTGRs have a relatively low power density compared with light water reactors, of the order of 2–3 MW m^{-3}. They include a large volume of graphite as moderator that also implies a relatively large core size. The core is usually annular to give a flat radial power distribution. HTGRs typically include a central graphite reflector and radial and axial reflectors, and are designed such that the inner reflectors that absorb a large fluence are replaceable. HTGRs exhibit good neutron economy due to the low absorption of the graphite and negligible absorption by the helium coolant. Another desirable feature is a negative reactivity core temperature coefficient that increases in magnitude at higher burn-up and lower fuel enrichment.

In current PBMR designs, the control rods for both operation and safety purposes are situated outside the reflector region in order to limit exposure at high temperature. This means that they have reduced worth, which tends to imply smaller diameter annular cores are designed. The fuel inventory is relatively low due to the use of low enriched fuel, which means that safety is not compromised. The power can also be effectively managed by varying the helium inventory and taking advantage of the negative temperature coefficient in the 25–100% power range.

HTGR core physics tools have been validated by comparison with the HTR-10 reactor in China, the high temperature test reactor (HTTR) reactor in Japan and the Proteus critical facility in Switzerland. Reactor physics methods have been applied utilising methods ranging from detailed Monte Carlo methods to combinations of cell transport and core diffusion models. Benchmarks have shown that some of these codes predicted the core criticality loading to a good level of accuracy. Thus, there are adequate methods available for reactor physics calculations for low-enriched gas-cooled reactors.

Regarding their thermal design, the characteristics features of HTGRs include low power density, high core thermal capacity with very high core outlet temperatures as high as 950°C, much higher than other reactor types. Other geometric features include a large height to diameter annular core with a steel pressure vessel, which enable decay heat removal under normal and abnormal conditions.

Modern designs utilise helium gas enabling a direct Brayton cycle to improve thermal efficiency and economics. The coolant circuit is based on gas at high pressure in the core, moving upwards to a gas plenum, cooling the external reflector regions and the upper core structures before entering the core flowing downwards. The gas then exits at temperatures in the range 800–950°C. Efficiencies of up to 50% are the target. More ambitious future designs have even higher temperatures as described below.

The power conversion unit converts the core thermal energy into mechanical and then electrical energy by means of various engineering components designed to achieve high efficiency. The gas turbine is connected to the generator, turbo-compressors to pressurise the helium, pre-cooler, inter-cooler and recuperator.

Different HTGR designers have proposed different direct and also indirect cycle designs. In the former case, the reactor vessel is connected by a cross-duct to the power conversion unit. In the latter case, primary and secondary circuits are interfaced by an intermediate heat exchanger (IHX). An advantage of the latter is to include an additional barrier against radioactive contamination of the turbo machinery. There have been considerable advances in turbo-machinery technology that have been achieved in parallel with the development of the Brayton cycle.

Below are briefly described some of the currently proposed designs of high-temperature thermal reactors. These are listed in Table 12.3.

12.3.1 VHTR (Gen IV)

The very high temperature reactor (VHTR) has been put forward by the GIF members as part of their Generation IV programme (The US Generation IV Implementation Strategy, 2003; Figure 12.2). This could be used for high efficiency electricity production but is also

Table 12.3. High temperature gas reactors

Reactor	Rating (MWe)	Country
VHTR (Gen IV)	~300	GIF members
GT-MHR	293	US/Russia/France/Japan
PBMR	120	South Africa/Consortium

Data from IEA/OECD (NEA)/IAEA (2002), The US Generation IV Implementation Strategy (2003), Squarer *et al.* (2001) and 18th Meeting of the Technical Working Group on Gas Cooled Reactors.

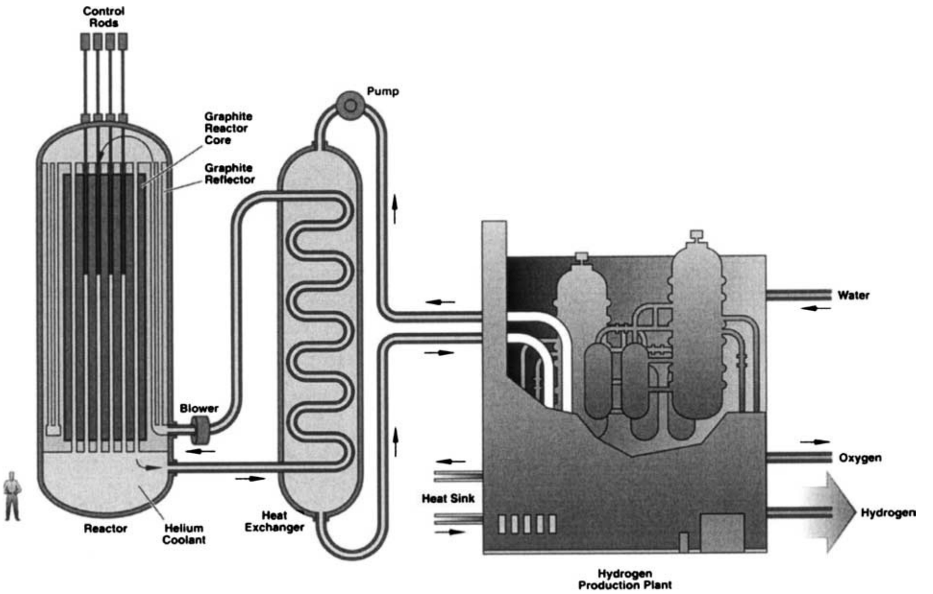

Figure 12.2. Very high temperature reactor. Source: NEA Annual Report (2002).

seen as a good candidate for hydrogen production by thermochemical water splitting or through high-temperature steam electrolysis. The reference reactor is a 600 MWt helium-cooled reactor with a coolant outlet temperature of 1000°C or more with a design efficiency of 50%. At this efficiency, it could produce 200 metric tonnes of hydrogen per day. This technology requires advances in high-temperature materials, alloys, ceramics and composite materials.

The VHTR is seen as a natural development of the gas turbine modular helium reactor (GT-MHR) and PBMR reactor designs out lined below. These designs are the latest in evolutionary high-temperature reactor technology that are being proposed for short- to medium-term deployment.

12.3.2 GT-MHR

This design has been put forward by General Atomics within their GT-MHR development programme as a future plant to produce electricity at high efficiency. It satisfies Generation IV objectives, having passive safety, good economics, improved proliferation resistance and better environmental attributes than the current generation of nuclear plants, in that it has better fuel utilisation and produces less waste. It has a high outlet temperature of 850°C and therefore has the additional potential for hydrogen production via high-temperature electrolysis or water splitting. The technology could be put forward for development within the next generation nuclear plant (NGNP) demonstration project at Idaho national engineering and environmental laboratory (INEEL). The nominal power for a single unit is envisaged to be about 293 MWe. The timescale for a possible construction would not be until about 2009. In regard to the economics, the overnight capital costs for 4 standard units is foreseen as about $1000 per kWe with 20 year levelled generation costs of 3.1 cents per kWh (based on 2003 dollars).

12.3.3 PBMR

The PBMR design has been put forward by the South African Utility, ESKOM in partnership with an international consortium. It also meets Generation IV design objectives in that it includes passive safety features to meet public acceptance criteria and offers competitive economics. The units are relatively small at 110–120 MWe with good economic and safety characteristics. The PBMR is also flexible in that it can be built virtually anywhere. It operates with a direct Brayton thermo-dynamic cycle, with target efficiency of around 45%. In principle, it can also use a thorium fuel cycle as well as a traditional uranium cycle. The design is modular in order to enable an operating utility to match the size of his station to the demand. The present capital cost is estimated at about $1000US per kWe, the construction period is estimated to be very short at around 2 years.

The PBMR offers a potential complementary service to the energy market in terms of present plant capabilities as both an electrical and non-electrical energy generator. It is of medium size, comparable with current-sized gas plants. It could offer a capability for the co-generation of heat or even dedicated nuclear heating applications, as expanded below.

The PBMR design is based on the HTR-MODULE design previously licensed in Germany for commercial operation. Present activities are aimed at the engineering design, independent safety reviews by participating countries in the ESKOM project and in making provisions for the licensing process.

The HTGRs have desirable features from various safety perspectives. The cores have a large thermal inertia, low power density and a strongly negative Doppler reactivity coefficient. As for most reactor types, the transients can be categorised into two broad categories, reactivity-initiated events and loss of flow events, either with or without depressurisation. For an un-scrammed core heat-up, the maximum core temperatures are reached within 3 days but fuel temperatures do not exceed above 1600°C ensuring that fuel particle integrity is preserved.

One concern with HTGRs is that air could ingress the core resulting in oxidisation of the graphite. This would require a multiple failure scenario of ruptures in the pressure vessel and surrounding concrete. However as noted above, even if such events occurred, there would still be several days to breach the opening of the reactor vessel.

A considerable advantage of gas systems described above is that they are free from the usual problems associated with loss of cooling in LWR systems. Thus there are no phenomena of concern such as 'Departure from Nucleate Boiling' loss of heat transfer or 'Pellet Clad Interaction' failure.

The reactor has diverse and redundant safety systems. For example, the reactor can be shutdown by three independent control systems. Each system is sufficient in itself to achieve this requirement.

In summary, in addition to electricity generation, HTGRs are being proposed as candidate plants for process heat applications that require high-temperature conditions. These include hydrogen and methanol production in a steam reformer, a process that requires high-temperature heating of steam and methane. Steam could be produced and then utilised for processes such as coal densification and steam injection for the recovery of hydrocarbons. These plants would also be suitable for de-salination processes, which require low-temperature heat. There may be potential to take waste heat from the pre-cooler that would otherwise be wasted. The ways of operating HTGRs in these multi-generation modes would add significantly to the thermal efficiencies that would be achievable with the plant.

12.4. GAS-COOLED FAST REACTOR

The technologies that have been investigated for a gas-cooled fast reactor concept have been reviewed by Mitchell *et al.* (2002). The basic idea has been to extend the thermal reactor concept but with a fast reactor core. The existing technology gas-cooled breeder reactor (ETGBR) was based on an AGR technology with a carbon dioxide-cooled system. The gas-cooled fast reactor (GCFR) design of General Atomics takes a helium technology as its basis. The gas breeder reactor (GBR) covered four different design concepts in respect of carbon dioxide and helium as possible coolants, oxide pins vs. particle fuel, etc., see below. These are surveyed in Table 12.4.

12.4.1 GFR (Gen IV)

The GCFR System is considered within the GIF programme (The US Generation IV Implementation Strategy, 2003; Figure 12.3). It is a good candidate for electricity

Table 12.4. Gas-cooled fast reactors

Reactor	Rating (MWe)	Country
GFR (Gen IV)	288	GIF members
ETGBR (old concept)	1320	UK
GCFR (old concept)	375	US
GBR 1–4 (old concept)	1000–1200	Europe

Data from The US Generation IV Implementation Strategy (2003) and Squarer *et al.* (2001).

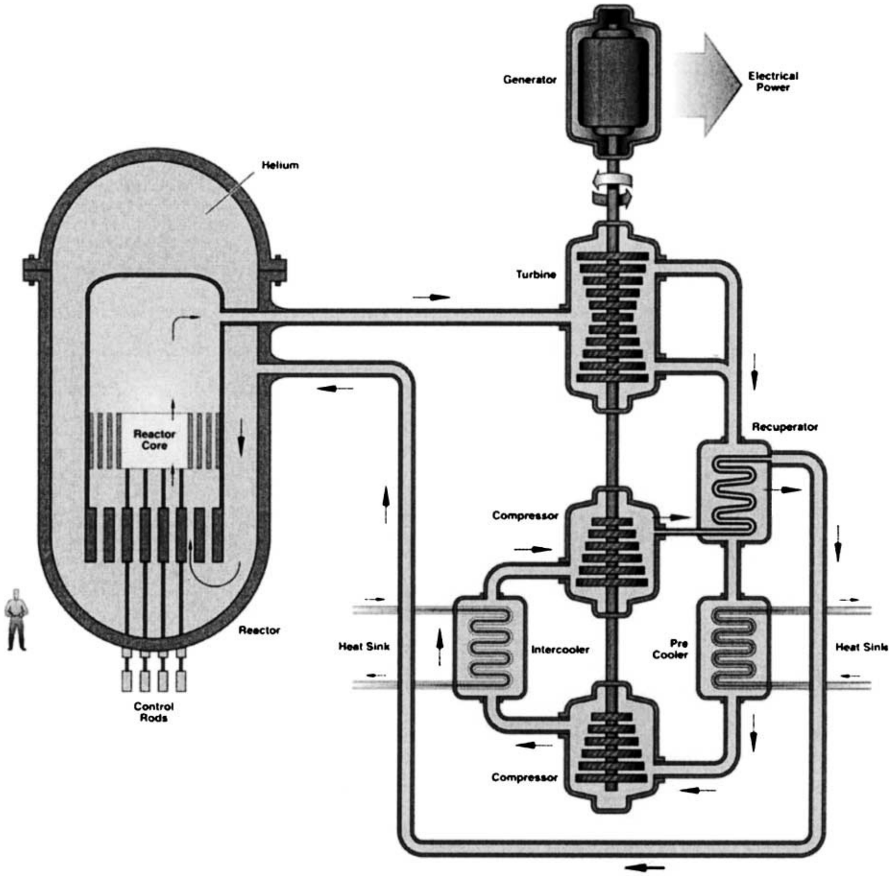

Figure 12.3. Gas-cooled fast reactor. Source: NEA Annual Report (2002).

production as well as actinide management. It may also be a candidate for hydrogen production. It is based on a closed fuel cycle and therefore offers a more sustainable fuel cycle option. The reference plant is 288 MWe. However, it requires significant advances in fuel and materials.

The GFR is a logical progression from VHTR gas-cooled systems described above and earlier European and US gas reactor technology. It is therefore seen as a longer term deployment option.

12.4.2 ETGBR

The ETGBR grew out of the AGR technology developed in the UK during the 1970s. The core design took advantage of lessons learned from both AGR technology and the fuel design took advantage of experience of the sodium-cooled fast reactor (SFR) experience.

An important objective for the fuel and core design was to obtain a good breeding gain. The fuel consisted of MOX or UOX in a steel clad. Reactivity was controlled by three diverse and separate control rod systems for both control and shutdown. The burn-up target was 10%.

The reactor coolant system consisted of an integrated AGR design, with the boiler and circulators contained in a pre-stressed concrete pressure vessel. The main components were based on AGR technology in terms of materials and design. Since the core temperatures were cooler than those for AGRs (limited by maximum clad temperature), the cooling system had to withstand a less demanding environment than for AGRs.

The containment was designed to be less embracing compared with the primary/secondary containment adopted for LWRs. It was, however, vented to mitigate the release under severe accident conditions.

It was felt that the design and safety philosophy of the ETGBR could be potentially licensable in the UK. There has also been recent interest in the design because it can be flexible in its fuel cycle. This could provide the option of achieving modest breeding or alternatively to enable the burning of plutonium and minor actinides.

The cost was reviewed in the 1970s, being found to be 10% greater than a PWR of the day. These costs were favourable in comparison with the AGR figure (increase of 25%) and the LMFBR figure of 60%. More recent studies have shown that the ETGBR could be economically competitive in comparison with other advanced reactors.

There have also been interest in GCFR technology in the US and a number of designs developed.

12.4.3 GCFR

The GCFR programme had taken place in the US from early in the country's history on nuclear power. It was considered in parallel to the US LMFBR programme, being perceived to offer a number of advantages. The conceptual GCFR was in principle simpler to operate compared with the sodium-cooled LMFBR and, if required, has the potential for higher breeding gain. Considerable experience had also accrued from the operation of the HTR Peach Bottom and Fort St Vrain reactors.

Regarding the fuel and core design, the GCFR was based on the Liquid Metal Fast Reactor (LMFR) design, incorporating niobium stabilised stainless steel pins and wrappers. The coolant was helium and the pin design was based on pressure equalised vented pins. This has the advantage for pin design and performance at the expense of a removal of a fission product release barrier. Reactivity was controlled by two independent and diverse shutdown systems.

The primary heat removal system was an upward flow system through the core, driven by active circulators. The design also included an independent and redundant decay heat removal system. The reactor vessel was engineered from pre-stressed concrete. On the helium side, it was insulated and there was water cooling on the outside.

The containment building incorporated a molten fuel containment system, below the bottom of the vessel.

The GCFR programme was halted by the USDOE in 1981, because Liquid Metal Fast Breeder Reactor (LMFBR) technology had progressed sufficiently to become a credible option. The GCFR had no proliferation advantages over the LMFBR design. There were also some economic, safety and technical factors that affected the decision to go forward. The multi-cavity pre-stressed concrete posed problems for manufacture and inspection and also problems for extrapolation. From a safety perspective, vented pins implied that the first barrier for fission product release was lost. Finally, there were concerns over spent fuel assembly cooling. Nevertheless, the initial work over the first few decades was viewed as providing a positive story in terms of technical design development and the safety and licensing activities that were carried out.

12.4.4 GBR

Four different design concepts were also investigated in the early 1970s by the European GBR Association, taking the LMFBR fuel and core technology with a gas thermal reactor system. These designs were not developed commercially at the time, due to their cost, and the preference for LMFR and LWR systems, which were further developed. They are now being reconsidered in modular designs with more favourable economics and with more natural safety characteristics.

Three 1000 MWe systems were considered, GBR1 (helium (He) and fuel pins), GBR2 (He and coated particles), GBR3 (CO_2 and coated particles) and one 1200 MWe system GBR4 (He and fuel pins). The latter was the favoured option at the time.

GBR4 had vented pins containing coated particle fuel. The coolant pressure was 90 bars, necessary to achieve the efficiency using fuel pins. Two independent shutdown systems were employed. There is an advantageous safety feature associated with a negative reactivity expansion coefficient. C&I systems were based on 1970s technology and therefore a new design concept today would require a more up-to-date approach.

The reactor pressure vessel enclosed an integrated system, incorporating the boilers in individual pods in the vessel in a design similar to that employed by the AGRs. An independent decay heat removal system was also included. The containment included an inner steel liner and outer concrete shell.

Additional safety features were incorporated to accommodate cooling in a depressurisation accident, un-tripped loss of flow, etc. At the time, it was concluded that further development was required in the plant safety concept, particularly in the field of severe accidents, core melt and containment.

12.5. SODIUM-COOLED FAST REACTORS

A significant amount of experience has accumulated from liquid metal (particularly sodium) cooled fast reactor operation. Twenty LMFRs, developed over the last 50 years, have been constructed and operated, resulting in nearly 310 reactor-years of operation (IAEA-TECDOC-1289, 2002). These include major large-scale prototype and demonstration LMFRs and experimental fast flux test reactors.

Fast reactor development is being delayed in countries with relatively slow energy consumption growth and significant fossil fuel resources. However in some countries, with more rapid growth, or with limited uranium or fossil fuel resource, there is still interest in fast reactors for power generation. There is also a more general interest in fast reactors for plutonium burning, minor actinide transmutation and also for non-power producing nuclear heat applications. The latter topics are considered in separate chapters later in the book.

The countries where there is still a significant development programme in LMFRs include France, India, Japan and the Russian Federation. Other countries including Korea and China also have an interest in LMFRs.

In this chapter, some of the proposed designs that are being considered are summarised (Table 12.5). The designs that meet the more stringent safety requirements and likely to be competitive against LWRs for energy generation include the European Fast Reactor (EFR), the Prototype Fast Breeder Reactor (PFBR) from India, the Demonstration Fast Breeder Reactor (DFBR) from Japan and the BN-800 from the Russian Federation.

12.5.1 SFR (Gen IV)

The SFR systems are part of the GIF initiative. These are envisaged in two scales. The large one would incorporate a MOX fuel, supported by a reprocessing system serving a number of reactors. The second would be based on a Pu-minor actinide–zirconium metal alloy fuel, developed within a pyro-metallurgical process in co-located facilities (The US Generation IV Implementation Strategy, 2003). The outlet temperature would be

Table 12.5. Sodium-cooled fast reactors

Reactor	Rating (MWe)	Country
SFR (Gen IV med-large)	500–1500	GIF members
SFR (Gen IV small-med)	150–500	GIF members
EFR	1500	EU consortium
BN-800	800	Russia
DFBR	660	Japan
PFBR	500	India

Data from The US Generation IV Implementation Strategy (2003), Methnani (2003) and IAEA-TECDOC-1083 (1999).

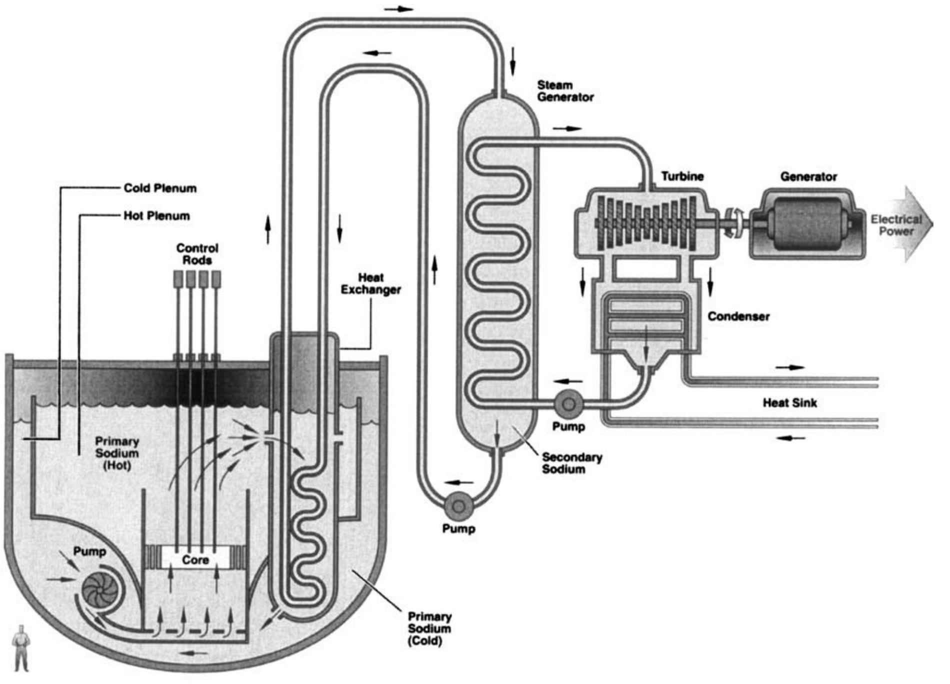

Figure 12.4. Sodium-cooled fast reactor. Source: NEA Annual Report (2002).

about 550°C. This system would mainly be used for electricity production or actinide burning. It is based on a closed fuel cycle, which is advantageous for actinide management. Both pool and loop type systems, the former shown in Figure 12.4, are possibly envisaged (Lennox, 2004).

The SFR concept takes advantage of the substantial expertise that has built up over many years of fast reactor operation in France, UK, US and elsewhere.

12.5.2 EFR

The EFR design has been completed which aimed to encapsulate the combined experience of France, Germany and the UK for liquid metal reactor technology based on pool-type reactors. Although construction is not foreseen in the near future, there is a design now available based on established technology and with realistic cost estimates.

The EFR project was launched in 1988 by the European Fast Reactor Utilities Group (EFRUG) including EdF (France), ENEL (Italy), Nuclear Electric (UK), Bayernwerk, Preussen Elektra and RWE (Germany) and BNFL (UK) and UNESA (Spain) joined later in 1993. Other design and construction companies 'EFR Associates' were also involved together with R&D companies to perform supporting experimental and theoretical studies.

The design objectives' lifetime were for high availability over a lifetime of 40 years. The technology was therefore based as far as possible on proven methods or methods that would be expected to be fully endorsed by appropriate R&D.

The reactor core consists of three radial core zones, with different plutonium contents with the inner, intermediate and outer zones with 207, 108 and 72 fuel assemblies, respectively, in a hexagonal lattice. Surrounding the core are 78 breeder subassemblies. Further, two options for the core design are possible, a homogeneous core and an axially heterogeneous core with axial breeder blankets. There are 24 control and shutdown rods and 9 diverse shutdown rods for fast shutdown.

Each fuel assembly has a bundle of 331 fuel pins and the breeder subassemblies have 169 pins. The fuel and the fertile material consist of pellets of UO_2 and $(U, Pu)O_2$, respectively. The control and shutdown rods are each retained in a hexagonal bundle of 37 absorber pins and the diverse shutdown rods each contain 55 absorber pins. These include B_4C absorber material.

The reactor and its cooling systems were based on a six circuit sodium coolant design. The reactor unit is an evolution of the Superphénix design. Sodium is circulated through the core region by three primary pumps. The heat is transferred to the secondary sodium loop by six IHXs. Each secondary loop transfers heat to a steam generator unit.

The safety concept is based on the 'defence-in-depth' approach. The system is at low pressure and loss of coolant accidents are precluded within the design basis. The prevention is based on enhanced shutdown and removal of decay heat. Decay heat removal is normally via the steam/water plant; there are in addition two diverse decay heat removal systems. An objective is to choose a core height to minimise the sodium voiding positive reactivity effect. Reactor shutdown is assured via two independent and diverse shutdown systems.

12.5.3 BN-800

Considerable Russian experience has been built up from operation of a number of experimental and prototype fast reactors including BR-10, BOR-60, BN-350 and BN-600 (IAEA-TECDOC-1289, 2002). BN-600 has operated reliably at Belojarskl in Russia (IAEA-TECDOC-1083, 1999). BN-600 has a nominal power output of 600 MWe and has been in operation in 1980 with an average load factor of 70% (IAEA-TECDOC-1289, 2002).

The major emphasis in the Russian Federation is in continued design improvement and improved economics. The main applications are seen to be for energy production and the conversion of plutonium and minor actinides. The design of BN-800 has been completed and a site licence issued for the construction of a BN-800 at Yuzno-Uralskya and Beloyarskaya in Russia (IAEA-TECDOC-1289, 2002).

BN-800 is based on a three circuit flow system incorporating three primary loops, three secondary loops and three steam generators. The reactor core and radial blanket are built up with assemblies in a hexagonal lattice. The fuel is MOX sintered pellets. Compared with BN-600, there are a number of design improvements.

These include improved economic and operational performance, enhanced reliability of components and simplification of systems (e.g. single steam turbine compared with three in BN-600). Improved safety is included with the introduction of a passive emergency protection system and improved safety system redundancy and diversity.

12.5.4 DFBR

The fast reactor programme in Japan is seen as part of their national nuclear fuel recycling programme (IAEA-TECDOC-1289, 2002; IAEA-TECDOC-1083, 1999). The FBR is under consideration to become the future long-term alternative to LWRs (IAEA-TEC-DOC-1083, 1999). Experience has been assimilated on fast reactor performance over the past 20 years. The experimental fast reactor 'Joyo', has been in operation during this period with good performance. The Power Reactor and Nuclear Fuel Development Corporation (PNC) also built a 280 MWe prototype fast reactor 'Monju' that operated until it was shut down in 1995 due to leak in the non-radioactive secondary circuit. The experience is being taken into account in the design of the 600 MWe DFBR, currently in progress.

The DFBR design includes a top-entry loop style arrangement. The primary circuit consists of a reactor vessel, three IHX vessels and three pump vessels. The secondary side is made up of a secondary pump, a once-through steam generator and connecting pipes on each loop. There are two shutdown systems.

The fuel is a Pu–U mixed oxide fuel and the core has two homogeneous regions with different plutonium enrichments. The core is designed for two phases of operation, an initial phase to produce an average burn-up of 90,000 MWd t^{-1} over a 13 to 15 month period with breeding radios of 1.2 or 1.05, with and without a radial blanket, respectively. An average burn-up of 15,000 MWd t^{-1} is envisaged for the high burn-up phase over an operating cycle of 20 months.

Research and development of the DFBR is via a collaboration between the PNC, the Japan Atomic Energy Research Institute (JAERI), the Central Research Institute of Electric Power Industry (CRIEPI) and the Japan Atomic Power Company (JAPC).

12.5.5 PFBR

The PFBR is a pool-type sodium cooled reactor under design in India (IAEA-TEC-DOC-1083, 1999). It is a 500 MWe medium-sized reactor and extrapolates from the FBTR 13.3 MWe experimental reactor that has already been successfully commissioned.

The fuel consists of mixed plutonium and uranium oxides and depleted uranium is used as the blanket. The fuel region includes two zones of different plutonium oxide enrichment. Secondary side shielding is included in both the axial and radial directions.

There are nine primary control and safety rods for setting the power level and for shutting down the reactor. There are in addition three diverse safety rods.

The primary circuit consists of two pumps and four IHXs, with one IHX on either side of each pump. The secondary sodium system consists of two identical loops each comprising of two IHXs and three steam generator modules.

12.6. LEAD AND LEAD–BISMUTH COOLED FAST REACTORS

12.6.1 LFR (Gen IV)

Lead and lead–bismuth systems are being considered in the GIF programme (The US Generation IV Implementation Strategy, 2003; Figure 12.5). Examples are listed in Table 12.6.

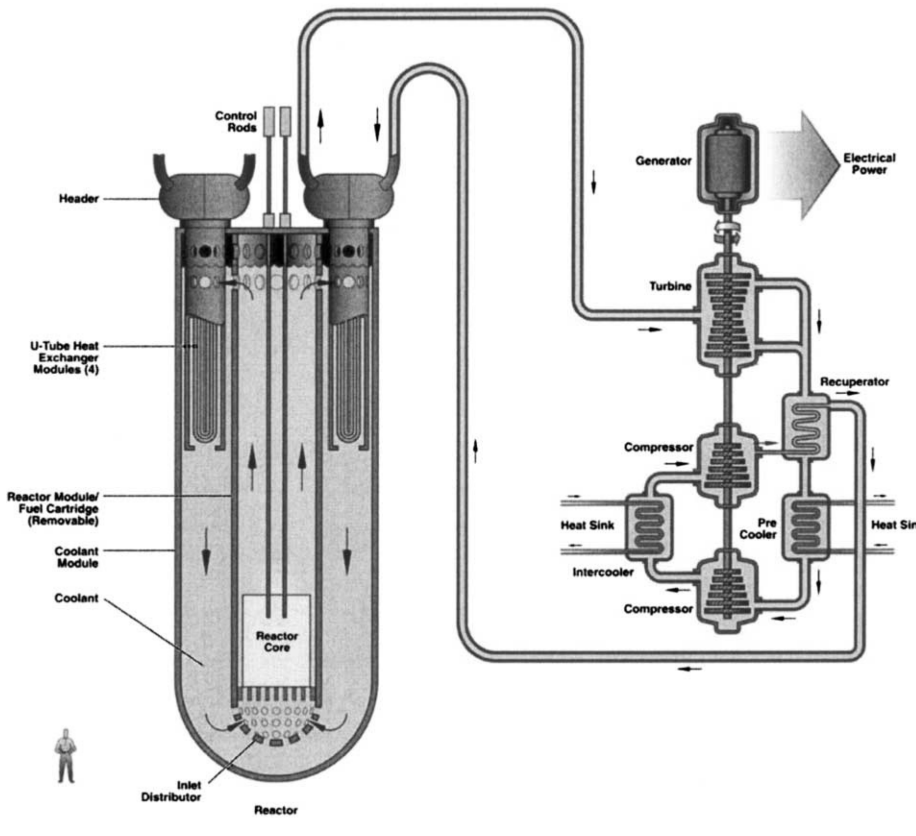

Figure 12.5. Lead-cooled fast reactor. Source: NEA Annual Report (2002).

Table 12.6. Lead and lead–bismuth cooled reactors

Reactor	Rating (MWe)	Country
Lead and lead–bismuth		
LFR (GEN IV)	50–1200	GIF members
Lead		
BREST-300/600	300/600	Russia
LCFR	1500 (MWt)	Japan
Lead–bismuth		
BRUS-150	150	Russia
SVBR-75	75	Russia
ANGSTREM	6–25	Russia

Data from The US Generation IV Implementation Strategy (2003) and IAEA-TECDOC-1289 (2002).

The system is based on natural convection cooling with outlet temperature 550°C. It could be somewhat higher ~800°C subject to improved materials development. It can be used within a long life closed fuel cycle of up to 30 years in some concepts. It is anticipated to be used for electricity production, hydrogen production and actinide management.

12.6.2 Lead Cooled

12.6.2.1 BREST-300. Lead cooled reactor systems are under study at the Institute of Physics and Power Engineering (IPPE) and the Kurchatov Institute (IAEA-TEC-DOC-1289, 2002).

In the BREST-300 designs, developed by RDIPE and Kurchatov there is a two circuit design, there are four parallel loops including pumped lead flow removing heat from the reactor core. Lead inlet and outlet temperatures are 420 and 540°C, respectively. The design is integral with a supercritical pressure (24.5 MPa) steam water cycle. The uranium and plutonium nitride fuel implies low moderation and absorption of neutrons hence it is possible to achieve a core breeding ratio equal to one.

The BREST-300 reactor has various safety features such as negative void temperature coefficient; it operates with a breeding ratio of near unity with consequently minimal excess reactivity and there are no soluble poisons in the reactor coolant (IEA/OECD (NEA)/IAEA, 2002). Regarding the coolant, there is decay heat removal by passive systems, increased reactor coolant inertia, and the system pressure is low.

It has good thermodynamic efficiency due to high core outlet temperature, reduced number of components in the nuclear steam plant and reduced containment design requirements. The relatively small size implies reduced capital cost and this together with increased core outlet temperature means that the plant is also applicable to process heat applications.

There are, however, some penalties in using lead. There is a much greater pressure drop (about 7 times greater than sodium) across the core for otherwise similar conditions of reactor power, coolant flow cross section area in the core and fuel element length. This is caused by the lower thermal capacity of lead compared with sodium. The higher density of lead compared with sodium does not compensate. Lead cooled reactors therefore need to have a reduction in fuel fraction and increase in core diameter to reduce the hydraulic resistance. This implies that the core dimensions of the BREST reactors are large.

12.6.2.2 BREST-600. The plant has been scaled up to 600 MWe by RDIPE in co-operation with RRC Kurchatov (IAEA-TECDOC-1289, 2002). The characteristics of BREST-300 and BREST-600 are similar.

There is also an active programme on lead cooled reactors in Japan.

12.6.2.3 LCFR. Design studies of lead cooled fast reactors (LFRs) with nitride have been performed by the Japanese (IAEA-TECDOC-1289, 2002) as part of their programme to improve uranium resource utilisation and for the transmutation of high-level waste nuclides. The Japanese studied the impact of plant size on seismic issues and ways of developing more compacted and integrated plant designs.

The LCFR has negative void reactivity but a high breeding ratio of 1.26. The design is integral with the core, support structure and primary heat exchange systems situated within the reactor vessel. On the secondary side, the once through steam generator and its helical tubes are situated around the core and core diagrid. Regarding safety characteristics, the design is to reduce the propensity for lead–steam interaction.

12.6.2.4 General. Lead cooled systems have both advantages and disadvantages compared with sodium systems. Lead is much less reactive with air and water compared with sodium. In terms of other advantages in regard to minimum reactivity excess, transmutation of old actinides and fission products, proliferation safeguards considerations, safety in accident situations and economic competitiveness, lead and sodium systems have comparable properties.

There are however some negative aspects of lead associated with its corrosiveness; it may freeze in the steam generator in the case of feed-water heater failure. Repair and maintenance and remote re-fuelling operations are carried out at high lead temperatures of over 400°C. There is a potential for fuel subassembly blockage caused by lead/water/steam interactions.

12.6.3 Lead–Bismuth Cooled

Considerable experience has been gained with lead-bismuth eutectic cooled reactors in the Russian Federation. This has been largely in connection with the development and operation of submarine propulsion reactors (IAEA-TECDOC-1289, 2002). Studies have been carried out by the Russian Federation Institute of Physics and Power Engineering (IPPE) and EDO Gidropress. Lead–bismuth offers some potential advantages, compared with lead, as a coolant and also some disadvantages; these are discussed below.

Lead–bismuth systems are being considered within the GIF Generation IV initiative. Several design concepts have been studied by the Russians. SVBR-75 is designed to produce 75 MWe and to operate for 10 years without refuelling. A smaller transportable combined heat and power version, ANGSTREM, has also been studied, generating 6 MWe. There is also a 25 MWe version being investigated by the Russians.

12.6.3.1 LFR (Gen IV). The main characteristics of the GIF reference design for lead cooled systems in general were discussed above.

12.6.3.2 BRUS-150. An integral type lead–bismuth reactor, generating 150 MWe, is being considered in the BRUS-150 project. All the lead–bismuth is contained in the reactor vessel, which contains the core, pumps and steam generators. This reactor is designed for the burning of weapons grade plutonium and the transmutation of minor actinides. In the present design of this (and other lead cooled) reactors, there is no intermediate circuit between the primary coolant and the water/steam secondary side. This is a concern in the event of steam generator leakage, which might result in Pb or Pb–Bi/water/steam interactions.

Pb–Bi and Pb share a number of similarities in terms of their thermal–hydraulic properties and also some advantages compared with sodium. For example, they have high boiling temperatures and relative chemical inertness compared with sodium. Pb–Bi has some advantages over Pb as a coolant in that it has a lower melting point (123.5°C) compared with Pb (327°C). A disadvantage in the use of Pb–Bi coolant is the formation of the volatile alpha emitter, polonium (^{210}Po) produced from bismuth (and to some extent from Pb). Therefore, leakage poses a hazard to the operators and to the environment in the event of a cover gas release. Careful chemistry control of the primary circuit is also required to avoid the formation of lead oxide and other impurities.

It is concluded in IAEA-TECDOC-1289 (2002) that there are problems in applying much of the experience and data gained from the Pb–Bi cooled submarine studies to commercial-sized lead cooled power plants. This is because of the much greater annual load factor required, the higher temperature of the lead primary circuit, and additional corrosion phenomena at the commercial plant scale. Thus, there is considerable R&D required to extrapolate from the lead–bismuth submarine experience to the civil commercial nuclear plant situation.

12.7. MOLTEN SALT REACTORS

Molten salt reactor (MSR) technology has been available since the 1960s. It was developed at Oak Ridge National Laboratory (ORNL) and the MSR Experiment (MSRE) which operated for nearly 3 years during the late 1960s (IEA/OECD (NEA)/IAEA, 2002). Examples are listed in Figure 12.6.

12.7.1 MSR (Gen IV)

The MSR is part of the Generation IV programme (The US Generation IV Implementation Strategy, 2003). In this design, the fuel is a liquid mixture of sodium, zirconium and uranium fluorides. The system is low pressure, with the coolant outlet temperature around 700°C. The power for the reference plant is 1000 MWe. It is a flexible system for actinide destruction. The economics are less favourable because of a large number of support systems for the maintenance of fuel and coolant. The system will require significant advances in chemistry plant design before it can realise a more mature design. The MS system will largely be for electricity production and plutonium and minor actinide destruction. Some example reactor types are given in Table 12.7.

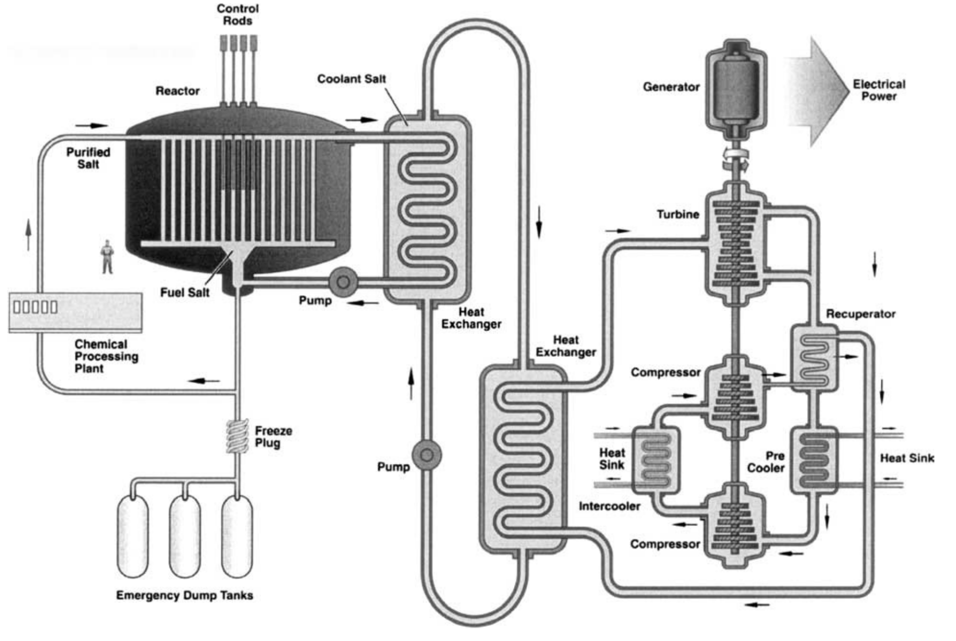

Figure 12.6. Molten salt reactor. Source: NEA Annual Report (2002).

Table 12.7. Molten salt reactors

Reactor	Rating (MWe)	Country
MSR (Gen IV)	1000	GIF Members
USR	625	US
MSR-NC	470	Russia
FUJI	100	Japan

Data from IEA/OECD (NEA)/IAEA (2002) and The US Generation IV Implementation Strategy (2003).

12.7.2 USR

The USR reference 625 MWe design developed by ORNL builds on earlier experience from the laboratory (IEA/OECD (NEA)/IAEA, 2002).

12.7.3 MSR-NC

The MSR-NC 470 MWe reference design has been put forward by the Kurchatov Institute (RRC-KI) in Russia.

12.7.4 FUJI

The FUJI reactor developed by ITHMSO in Japan is a low-pressure vessel loop style reactor with a graphite moderator and a molten salt coolant. It built on the ORNL technology and has an electrical output of 100 MWe. It includes inherent and passive features in that no moderating materials are located near the reactor vessel. Thus the reactor cannot achieve criticality outside of the core in the event of molten salt leakage.

In MSRs such as FUJI, the fuel (uranium and thorium) is dissolved in the molten salt. The salt is $^{7}LiF-BeF_2$ and it can contain fissile material, $^{233}UF_4$, and fertile material, $^{232}ThF_4$. The temperature reactivity coefficient is strongly negative with increasing temperature due to the presence of the graphite moderator and reduction in molten salt density.

The FUJI reactor has other important safety features, decay heat can be passed passively to the environment, on-line fuelling ensuring that reactivity is minimum at all times, the pressure is low and the vessel is designed against high fluence embrittlement. There are no soluble poisons in the molten salt coolant.

Economically, there is good thermodynamic efficiency because of the high core outlet temperature. This also makes the reactor a good candidate for combined heat and power applications compared with present generation water reactors. The reactor has a smaller number of components, reduced containment requirements and is of small size so capital costs are kept down. It has on-line refuelling so refuelling outages are eliminated.

Environmentally, this fuel cycle has some attractions; the presence of thorium implies that a smaller number of higher actinides are produced. Over the operating life of the reactor, the fuel is not removed, so no fission products are removed from the fuel/coolant. It also operates as a near breeder (breeding ratio near unity); thus uranium resource requirements are reduced.

REFERENCES

Dobashi, K., Oka, Y., Koshizuka, S. (1998) Conceptual design of a high temperature power reactor cooled and moderated by supercritical light water, 6th International Conference on Nuclear Engineering, ICONE6, ASME, NY.

Duffey, R.B. (2004a) The super critical water reactor: past, present and future, Generation IV Seminar on Nuclear Energy Systems Research and Development, Institute of Nuclear Engineers, DTI, London, March 2004.

Duffey R.B. (2004b) The role and future of nuclear energy: a Canadian perspective, Generation IV Seminar on Nuclear Energy Systems Research and Development, Institute of Nuclear Engineers, DTI, London, March 2004.

Generation IV Seminar on Nuclear Energy Systems Research and Development (2004) Institute of Nuclear Engineers, DTI, London, March 2004.

IAEA-TECDOC-1083 (1999) Status of liquid metal cooled fast reactor technology, Vienna.

IAEA-TECDOC-1289 (2002) Comparative assessment of thermophysical and thermodraulic characteristics of lead, lead–bismuth and sodium coolants for fast reactors, Vienna.

IEA/OECD (NEA)/IAEA (2002) Innovative nuclear reactor development: opportunities for international co-operation, 'Three Agency Study'.

Lennox, T. (2004) Generation IV sodium cooled fast reactor, Generation IV Seminar on Nuclear Energy Systems Research and Development (2004), Institute of Nuclear Engineers, DTI, London, March 2004.

Methnani, M. (2003) Status of high temperature gas-cooled reactor technology, IAEA HTGR Workshop ICTP Centre, Trieste, Italy, July 7–11, 2003.

18th Meeting of the Technical Working Group on Gas Cooled Reactors, Istanbul, High temperature gas-cooled technology, http://www.iaea.org/programmes.

Mitchell, C.H., Farrar, B., Lo Pinto, P., Dominguez, M.T., van Heek, A., Every, D. & Carluec, B. (2002) Gas-cooled fast reactor concept review studies (GCFR), Innovative nuclear reactor development: opportunities for international co-operation, IEA/OECD (NEA)/IAEA 'Three Agency Study'.

NEA Annual Report (2002) Trends in nuclear power, http://www.nea.fr/html/pub/activities/ar2002/nea-ar-2002-trends.pdf.

Proceedings of the First International Symposium on Supercritical Water-Cooled Reactors (2000) Design and Technology, SCR-2000, The University of Tokyo, Tokyo, November 2000.

Silin, V.A., Voznesensky, V.A. & Afrov, A.M. (1993) The light water integral reactor with natural circulation of the coolant at supercritical pressure B-500 SKDI. *Nucl. Engng. Design*, **144** (2), 327–336.

Squarer, D., Oka, Y., Bittermann, D., Aksan, N., Matraczy, C., Kyrki-Rajamaki, R., Souyri, A. & Dumaz P. (2001) High performance light water reactor (HPLWR), FISA: EU Research in Reactor Safety, EUR 20281, November 2001.

The US Generation IV Implementation Strategy (2003) US Department of Energy Office of Nuclear Energy, Science and Technology, http://gen-iv.ne.doe.gov, September 2003.

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Chapter 13
Accelerator Driven Systems

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Chapter 13

Accelerator Driven Systems

13.1. INTRODUCTION/OBJECTIVES

Innovative accelerator driven systems (ADS) are under study nationally and internationally to provide a possible alternative to critical reactor systems (considered so far in this book). The purpose of this chapter is to examine this technology. ADS are possible candidates for a range of applications; importantly, they provide a means of separating and eliminating actinides by a process referred to as partitioning and transmutation. They can transmute long-lived radioisotopes into short-lived or even non-radioactive isotopes, using an excess of neutrons available from a fission chain reaction. These neutrons are generated in a hybrid sub-critical reactor accelerator system, which forms the basis of the ADS. In such a system, high-energy protons produced by an accelerator bombard a 'target', producing an intense neutron source; this part of the process is termed 'spallation'. These neutrons are multiplied up in a sub-critical reactor, referred to as the 'blanket', which surrounds the spallation target.

Thus, an important application for ADS technology could be to transmute high-level nuclear waste to non-radioactive materials or materials with much shorter half-lives. The issue of highly radioactive waste produced from reactor operation is a continuing problem with regard to the future of nuclear power. Such waste must be managed in a safe and efficient manner if nuclear power is to be sustained in the modern world. Other ADS applications include the 'burning' of weapons grade plutonium and energy production. These are developed further below.

In regard to international activity, the IAEA compiled a status report in 1997 (IAEA-TECDOC-985, 1997a), requested by participants in a special scientific programme initiated in 1994 on the 'Use of High Energy Accelerators for the Transmutation of Actinides and Power Production'. The objectives were to review the various technical options available, including their advantages and disadvantages, including technical and economic viability and the future role of IAEA in developing international collaboration.

The process of nuclear transmutation has existed for some time. In 1919, Rutherford demonstrated the process for lighter elements, and Laurence in the US and Semenov in the USSR made attempts to promote accelerators to generate neutron sources in the 1940s. The evolution progressed through attempts to achieve transmutation using only spallation neutrons but these suffered from technical limitations and inefficiency. In recent years, hybrid systems have been produced involving the combination of a sub-critical reactor with a high-energy particle accelerator.

A number of different systems have been and are still being proposed. These include ADS using fast neutrons for higher actinide incineration; such systems have been proposed in the US and Japan. In the US, hybrid systems, using thermal neutrons with a linear accelerator has also been considered. Collectively the different approaches provide a means for the incineration of plutonium, for the transmutation of higher actinides and long-lived fission products (LLFP) to reduce radioactive waste activity, and for potential energy production using thorium fuel. In Europe, nuclear energy production from thorium-based fuel via the 'Rubbia' system has been put forward. The thorium fuel option reduces the concern about higher actinides in used fuel, and utilises relatively cheap and available thorium. These ideas have been tested using preliminary experiments at CERN.

ADS have the inherent safety feature that they are based on a non-self-sustained chain-reaction. This improves the safety characteristics of ADS and can also reduce or eliminate the need for control rods. A section on safety is included later in this chapter.

13.2. PHYSICAL SYSTEMS

ADS can be classified into a number of different systems, depending on the nuclear energy spectrum, the form of fuel (solid or liquid), fuel cycle and the coolant/moderator type. An important ADS design requirement for all systems and applications is inherent sub-criticality, reactivity stability and good neutron economy, the latter determining the power and cost. A summary of the various applications is given in Table 13.1.

Various fast and thermal neutron systems have been considered for both solid and liquid fuels and utilising different coolants, depending on the application. Systems that aim to take advantage of intermediate neutron resonances are also being considered, as are quasi-liquid fuel forms and other variations. Examples of some of the many different concepts are given in the next section.

Concerning accelerators, the Linac driven (linear accelerator) has been favoured for most concepts but the cyclotron-driven accelerator is also being considered in some groups (e.g. CERN). These are also addressed later in the chapter.

Table 13.1. Applications and benefits of ADS

Transmutation of nuclear waste and reduction of long-lived radiological hazard
Utilisation of existing weapons grade plutonium for energy production
Consistent with proliferation resistant fuel cycle management
Utilisation of thorium resource for energy production
Operational flexibility, offering sub-critical mode of operation

IAEA-TECDOC-985 (1997a).

13.2.1 Summary of Different Physical Systems Being Developed

During the spallation process, the collision between the energetic particle and the target nucleus leads to direct reactions referred to as intra-nuclear cascade. In this cascade, small groups or individual nucleons (protons and neutrons) are expelled from the nucleus. At energies above a few GeV per nucleon, the nucleus can fragment. After the intra-nuclear cascade, the nucleus is in an excited state and subsequently releases 'evaporates' nucleons, mainly neutrons.

The spallation process is complicated and depends on the target thickness and the target materials. For thick targets high energy (>20 MeV) secondary particles may take part in further spallation reactions. For some target materials, low energy (<20 MeV) neutrons produced from cascade evaporation, can enhance neutron production. For heavier nuclei, high-energy fission may compete with evaporation. Examples of materials that undergo spallation/high-energy fission include lead, tantalum and tungsten. Some spallation target materials, e.g. thorium and depleted uranium may be fissioned by both high- and low-energy neutrons. Regarding target particles, deuterium and tritium produce more neutrons than protons in the below 1–2 GeV energy range but the low-energy part of the accelerator tends to get contaminated, resulting in higher maintenance costs.

The requirement for an ADS target is to convert a high-energy particle beam to neutrons at low energy. It is desirable for it to be of compact size, to couple to a surrounding blanket, operate in the 10–100 MW power range, and have high neutron production efficiency. Other requirements, in common with other nuclear devices, are that it should be reliable and of low cost, be safe and generate only a small amount of waste. Molten lead is a good choice for meeting these requirements. Lead–bismuth eutectic has also been considered because of its lower melting point than lead, but this eutectic produces polonium, the release of which may be a problem at high temperature.

The blanket (sub-critical assembly) surrounding the target multiplies the spallation neutrons for the transmutation of the minor actinides (MA) and LLFP. Taking account of many aspects, safety, operations, material cost and incinerator costs, k_{eff} values for the target in the range 0.9–0.98 are typically considered.

The different neutron spectrum modes have different advantages and disadvantages. The thermal cross-section for transmuting MA and fission products is larger than the fast neutron cross-section enabling core inventories to be reduced substantially, but the thermal neutron cross-section of the transmuted products is also large; so neutron capture is a problem. From the point of view of neutron economy, the fast reactor is better than the thermal reactor.

The Th–U fuel cycle is an attractive option for future ADS because it produces a relatively small amount of higher actinides compared with the U–Pu cycle. The Th–U cycle is safer from a weapons proliferation standpoint because of the existence of the hard gamma emitter in the ^{232}U decay chain and because ^{233}U can be diluted by depleted or natural uranium in the start-up or feed fuel. Against these advantages, the Th–U fuel cycle has a less favourable neutron balance.

13.3. FUEL CYCLES

Different fuel and fuel cycle concepts have been considered in the reactor, arranged in a sub-critical state. Some of the primary areas of research at various laboratories are shown in Table 13.2. These are expanded further in Section 13.8.

Brookhaven National Laboratory (BNL) has focused mainly on fast spectrum concepts, liquid sodium cooling and oxide or metal solid fuels based on sodium-cooled fast reactor technology. Also in that laboratory, particle bed/bead fuel has been investigated in the thermal spectrum, as considered in space propulsion reactor technology.

In Japan, the Japanese Atomic Energy Research Institute (JAERI) has concentrated on MA burning in a fast neutron spectrum, with solid fuel also based on sodium-cooled fast reactor technology, or molten chloride fuel, as yet an unproven technology.

Los Alamos National Laboratory (LANL) has developed a concept based on a thermal neutron spectrum with molten fluoride fuels with different fissile materials such as weapons grade plutonium, LWR spent fuel (minor actinide and fission products) and thorium fuels based on molten salt water reactor technology. Liquid lead–bismuth systems in the fast spectrum have also been considered.

Table 13.2. Fuel cycle concepts and applications

Laboratory	Spectrum	Fuel	Application
BNL	Fast	Solid U/Pu, Na/Pb cooled	Energy production/MA&FP incineration
	Thermal	Particle U/Pu, He cooled	MA&FP incineration
JAERI	Fast	Solid U/Pu, Na cooled	MA incineration
	Fast	Molten chloride salt, U/Pu	MA incineration
LANL	Fast	U/Pu, Pb–Bi cooled	MA incineration
	Thermal	Molten fluoride salt, U/Pu	Pu destruction/MA&FP incineration
	Thermal	Molten fluoride salt, Th/U	Energy production
CERN	Fast	Solid ThO_2/UO_2, Pb/Pb–Bi cooled	Energy production and waste transmutation
ITEP	Fast	U/Pu, molten fluoride or Pb/Pb–Bi cooled	Pu destruction/MA&FP incineration
	Thermal	Solid W–Pu, heavy water	Pu destruction
	Thermal	U/Pu, heavy water solutions	Energy/MA&FP transmutation
CEA	Fast	U/Pu, Pb cooled	MA incineration

IAEA-TECDOC-985 (1997a).

The CERN group in Geneva, Switzerland, has put forward the concept of solid $ThO_2/^{233}UO_2$ fuel in a fast spectrum based on liquid lead/liquid lead–bismuth reactor technology. This uses a cyclotron-based system. The applications are for energy production or waste transmutation.

At the Institute of Theoretical and Experimental Physics (ITEP) in Russia, different technologies for the conversion of weapons plutonium and long-lived radioactive waste are being considered. These include heavy water suspensions, molten fluoride and liquid lead fast spectrum systems.

Work is also being carried out in various laboratories within the EU, including France, Germany, Italy, Sweden and the UK. Within France for example, research, carried out at CEA has focused on options for radioactive waste management.

There are clearly many options under investigation in the international community, which offer a reprocessing capability for nuclear fuel and in the case of weapons plutonium, a means for the reduction in the world's stockpile of plutonium. Further assessment of the various options is continuing.

13.4. NEUTRONICS AND TRANSMUTATION

There are a number of issues impacting the choice of ADS neutronic parameters, in particular the ADS reactivity, k_{eff}. The degree of sub-criticality (k_{eff}) must be a balance between safety and acceptable economics. Here k_{eff} represents the sum of the initial reactivity and all other possible effects, e.g. burn-up reactivity swing including Np or Pa effects, power and void reactivity, etc.

ADS can be used for minimising the sources of long-term radiotoxicity, e.g. reactor fuel inventories, fuel wastes from reprocessing, and long-lived radioactive fission products (Slessarev, 1997). According to Salvatores *et al.* (1995), the latter two of these sources are the most important in terms of the accumulation of radiotoxicity.

For example, consider the neutronic potential of a representative ADS within a uranium fuel cycle complex (Slessarev, 1997) in the following system. A slightly sub-critical lead-cooled fast breeder reactor with nitride fuel and proton beam source with a k_{eff} of 0.98 would exhibit a neutron surplus of about 0.4 neutrons per fission (zero breeding gain in the fuel) plus 0.05 neutrons/fission due to spallation in the lead target. The lead is used as a liquid and target. This gives a total neutron surplus of 0.45 neutrons/fission, sufficient to burnout all dangerous fission products and/or reproduce new fuels for further nuclear power utilisation.

In this system, there is no need for control rods; it is a dual circuit, and a relatively inert coolant from the point of view of safety, e.g. fire hazard. The neutronics are sub-critical plus a stabilised reactivity increment. This system provides an apparently good balance with regard to economics, reduction of fuel waste potential and safety for the uranium fuel cycle.

The thorium fuel cycle has a much lower waste toxicity level for both thermal and fast reactors than does the uranium fuel cycle. This is because of its smaller production of trans-plutonium (Carminati *et al*., 1994; Rubbia *et al*., 1995) and, therefore lower minor actinide concentrations (at least for about 1000 years before some build up of long-term toxic ^{233}U, ^{234}U, ^{231}Pa). From a neutronic perspective, however, every fission of ^{231}Th produces fewer neutrons than does ^{238}U. There are other disadvantages in relation to achieving sub-criticality at economic cost and a protactinium effect, which implies a low k_{eff} value. Thus for the thorium cycle, it is necessary to have a compromise between the economics, sub-criticality level and safety margin. This is difficult because a low k_{eff} can only be achieved at more expense; reduced cost would be at the expense of higher k_{eff} and less safety margin.

13.5. ACCELERATORS

Accelerator technology has been developed over several decades and there is some confidence developed in the technology. There are several approaches. The attributes of the different systems are summarised in Table 13.3.

Linear accelerators or Linacs are thought to be achievable up to relatively high power (200 mA, 1.6 GeV). They have been demonstrated as reliable and efficient research tools, and can be made available at a reasonable cost. The most efficient operating conditions for a linear accelerator at the present time would be around 100 mA.

Cyclotron, i.e. circular proton accelerators' technology has also advanced enabling a 10–15 mA proton beam to be achievable via a segmented cyclotron or synchrotron concept. The most efficient operating current for these is around 10 mA. They have some benefits compared with a Linac but also some disadvantages. The cyclotron

Table 13.3. Accelerator driven systems

System	Attributes
Linear accelerator (Linacs)	Achieved a reliable and efficient status Order of magnitude higher beam power than cyclotron Performance and safety-related issues in splitting the beam, e.g. to drive several sub-critical reactors
Circular proton accelerators (segmented cyclotron or synchrotron)	Occupy a smaller physical area than Linacs Limitations on maximum beam current of cyclotron Multi-stage parallel cyclotron arrangements may offer some advantages

IAEA-TECDOC-985 (1997a).

occupies a smaller physical area and is cheaper than the Linac, but the space limitation limits the proton current, in the present day to about 10–20 mA. Linacs do not suffer this limitation.

On a larger commercial scale, one option might be to use one linear accelerator to a number of sub-critical reactors by splitting the beam. However, there may be drawbacks in the event of failure of the beam dividers, in which case the full beam might be directed against one target, or failure of the full beam would shut down all the sub-critical reactors.

This problem could be overcome by using one or more smaller cyclotrons, running several smaller reactors, but at increased cost. Regarding the status of cyclotron technology, cyclotrons of 1.1 MW beam power for a 600 MeV proton accelerator have been developed at the Paul Scherrer Institute (PSI). A number of alternative options are under consideration, e.g. a 'multi-stage-parallel' cyclotron arrangement in which several lower energy, low current cyclotrons input into a high-energy cyclotron. This approach would also give some cost benefits in terms of energy scaling, compared with a linear accelerator.

13.6. RADIATION EFFECTS

There are clearly significant areas of research required to realise the ADS technology. Some broad scope areas are given in Table 13.4. These relate to general requirements needed for most of the different fuel cycles and applications. There are also particular engineering-related materials issues associated with radiation damage, and the need to extend the methodologies developed for critical reactors to the more complicated ADS-coupled transport situation.

Severe radiation damage can occur as a consequence of high current, medium-energy protons being injected into the target (Takahashi and Gudowski, 1997). Neutrons and charged particles are generated at energies reaching those of the protons causing radiation damage to the target and surrounding structural materials. This stems from the

Table 13.4. Research requirements

Transmutation of commercial power plant waste, particularly reactor grade plutonium
Deployment of weapons grade plutonium in power production
Assurance of proliferation resistant fuel cycle
Benefits and utilisation of the thorium fuel cycle
Impact of different ADS options on radiotoxicity of the fuel cycle reduction
Materials-related research, e.g. radiation damage of the target regions
ADS safety issues and their resolution
Methodologies development for ADS, e.g. necessary developments of critical reactor models

IAEA-TECDOC-985 (1997a).

displacement of lattice atoms within the target and from the energy the atom receives following emission of a nuclear particle, e.g. γ ray (Wechsler *et al.*, 1995).

The primary concerns on the effects of damage relate to hardening and embrittlement and the changes in mechanical properties and stability. The embrittlement is characterised by radiation defect clusters, helium aggregation to form bubbles, ductile brittle transition effects, and impurities arising from transmutation products.

The areas of particular damage will be surrounding walls and the window, which therefore needs to be replaced frequently in high-energy accelerators. Thus, damage is likely to be worst for a high-power accelerator with a large sub-critical reactor. This may be mitigated by adopting a concept with a smaller current and smaller sub-criticality. Similarly the structural damage in an accelerator driven system might be expected to be higher than in a corresponding critical reactor (Takahashi *et al.*, 1994).

The adoption of suitable materials for the beam window section and the target side walls is a subject for research.

13.7. MODELLING

Neutron transport in the fission range of heavy metal energies has been studied for many years within the nuclear reactor industry. In ADS, the situation is more complicated than in conventional nuclear reactors. In this case, there is dual transport modelling required, the transport of medium energy charged particles in the energy range 1–3 GeV in the spallation target, and the transport of neutrons down to low-energy range.

A two-step process of spallation and evaporation of the residual nucleus occurs when medium-energy protons collide with a nucleus. If the residual nucleus has high mass and moderately high excitation energy, it might undergo fission in competition with the evaporation reaction.

In regard to presently developed methodologies, the nuclear cascade processes can be calculated by the NMTC (Coleman and Armstrong, 1970) and HETC (Radiation Shielding Information Centre, 1977) codes using two-body collision theory, which is valid until a particle slows down and its wavelength becomes longer than the average distance between the nuclei. In this regime, an optical potential model can be used, based on quantum mechanics. These codes have been developed to calculate high-energy fission, for targets with high atomic number such as uranium and the actinides, by various laboratories including JAERI (NMTC) (Nakahara and Tsutsui, 1982), BNL (NMTC) (Takahashi, 1984), LANL (LAHET) (Prael and Lichtenatein, 1989). Other nuclear cascade codes FLUKA (Ranft *et al.*, 1985) and CASIM (VanGinnekin *et al.*, 1971) have been developed by the international community.

Two areas of microscopic nuclear physics have been studied by OECD/NEA, using data from a thin target benchmark and transport modelling using thick target physics (IAEA-TECDOC-985, 1997a).

13.8. INTERNATIONAL PROJECTS

13.8.1 BNL

There are three types of ADS under study at BNL (Takahashi, 1997).

The accelerator driven energy producer (ADEP) (Bonnaue *et al.*, 1986) is intended for energy production, incineration of MA and LLFP. This concept uses a small power accelerator similar to that of a segmented cyclotron.

The concept is close to that of the conventional Pu fuelled fast reactor, but is run in slightly sub-critical conditions of k_{eff} equal to 0.98–0.99. The cyclotron with a few mA current and 3 GeV energy protons supplies a small spallation source.

The fuel in the ADEP core region is $^{239}Pu + {}^{238}U$, and MA in metal and oxide forms. The reactor has thorium oxide in the blanket region. For transmuting the LLFP such as ^{99}Tc and ^{129}I (by neutron capture), the moderator region is installed between the outer core and blanket. To increase the production of ^{233}U, the moderator region can be fuelled with thorium oxide.

The second approach, known as the Phoenix concept (Van Tuyle *et al.*, 1993) (Figure 13.1) has the purpose of transmuting large amounts of MAs and LLFPs. It is based on modules of accelerator driven sub-critical lattices containing minor actinide fuel. From 1–8 modules serve as a target for an expanded proton beam of power 104 mA of 1.6 GeV protons. Each module of the core has a k_{eff} of 0.9 and is based on the fast flux test facility (FFTF) approach, e.g. oxide fuel elements and sodium cooling; with this specification the power generated would be 3600 MW.

The third concept is the accelerator driven particle fuel transmutator (ADPF) (Takahashi, 1990) which also transmutes MAs and LLFPs but at higher rate. This is achieved by means of a high neutron flux via the use of particle fuel.

Particle fuel can be used to generate high thermal power densities, because of its large heat transfer area (Takahashi, 1990). Helium is taken as the coolant because of the data available from conventional HTR technology.

13.8.2 JAERI

A national programme OMEGA started in 1988 for R&D in new technologies for the partitioning and transmutation of high level waste (Takizuka, 1997). The OMEGA programme consists of two areas of research, the separation of elements from high-level waste based on their physical and chemical properties and the transmutation of MAs and LLFPs into short lived or stable nuclides. A conceptual design programme has been put

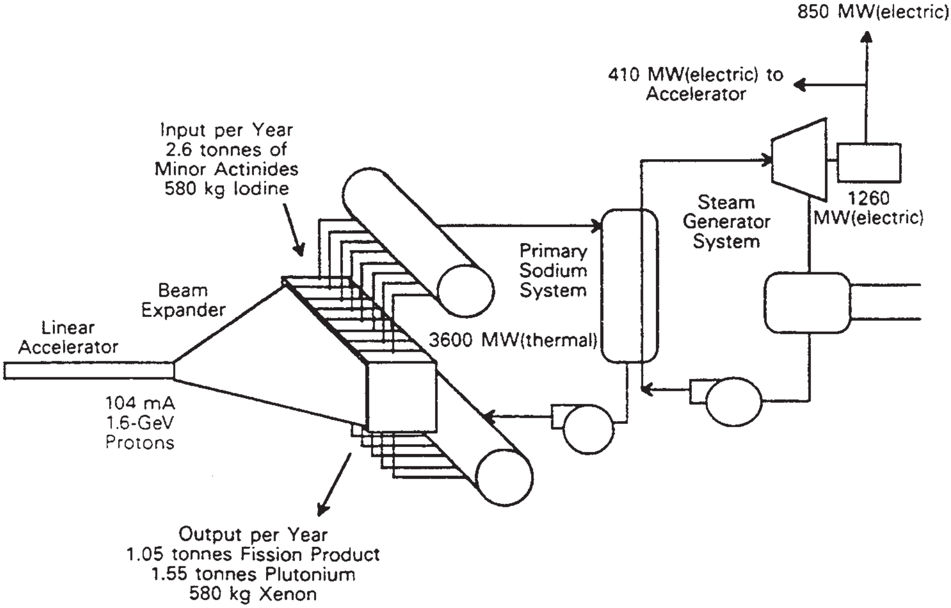

Figure 13.1. PHOENIX concept. Source: Van Tuyle *et al.* (1993).

together including code systems (Nakahara and Tsutsui, 1982; Nishida *et al.*, 1990) and integral experiments (Takada *et al.*, 1992) to investigate two concepts, a solid system and a molten salt system.

For the solid system, the design is based on a sodium-cooled fast reactor. The accelerator injects a 1.5 GeV proton beam onto a tungsten target, surrounded by a sub-critical blanket of actinide alloy fuel. The target blanket is at a total thermal power of 820 MW cooled by downward flowing sodium. The remainder of the heat transfer cycle is based on a tertiary cycle system (Figure 13.2).

The other design study is based on a molten salt target/blanket system, generating 800 MWt. The molten salt acts as a fuel and target and also as a coolant. The beam is 1.5 GeV. The latter concept is based on future generation reactor technology.

13.8.3 LANL

LANL has been studying accelerator driven transmutation technology (ADTT) for the destruction of nuclear waste and for generating power by systems which do not generate hazardous waste, and destroy their own waste. One particular approach called the accelerator driven energy production (ADEP) process, generates nuclear energy from thorium, avoids the production of plutonium and destroys its long-lived high-level fission product waste (Bowman, 1997).

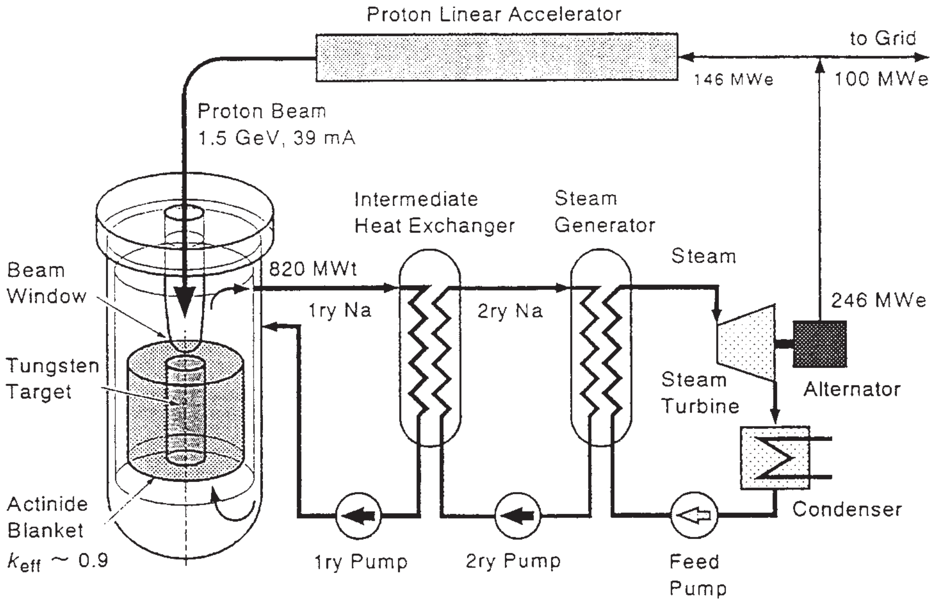

Figure 13.2. Sodium-cooled fast reactor concept of transmutation system. Source: Takizuka *et al.* (1992).

The system is based on ^{232}Th, which is converted by neutron absorption to the fissile component ^{233}U from which the energy is generated. The system contains a target/blanket that contains the fissile material and the waste to be destroyed. A continuous chain of fissions is produced, by an external neutron source that allows for the expenditure of neutrons on waste destruction (there are about 5–10% fewer neutrons than would otherwise be necessary to maintain a chain reaction). Thus without this external source, the system would not be self-sustaining.

In the ADTT, an 800 MeV proton beam is directed onto a lead target in an assembly containing the target and a surrounding blanket including the fissile material. The blanket acts as a moderator and consists mostly of graphite and molten salt containing the fissile fuel as an actinide fluoride. The graphite and molten salt are compatible; this has been established from long-term experience with the molten salt reactor at ORNL (Weinberg, 1970). The system multiplies the neutrons produced by the beam by about a factor of 20 operating at a k_{eff} of 0.95. Heat is removed from the blanket by internal heat exchangers, which transfer heat from the primary working salt to a secondary external salt stream, thence to a steam generator for electric power production. The majority of power generated goes to the grid; about 10–15% is used to power the accelerator. The liquid fuel system is continuously fuelled. It is regularly cleaned to enable fission products to be continuously removed.

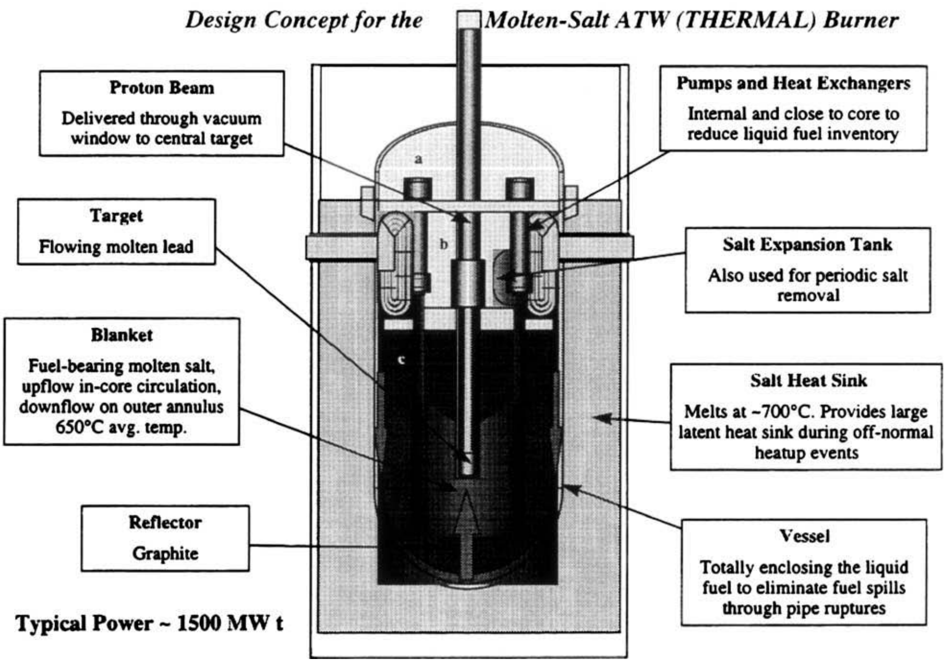

Figure 13.3. Conceptual design concept for a molten-salt ATW burner. Source: Cowell *et al*. (1995).

The accelerator transmutation of waste (ATW) project is part of the ADTT programme. It has the specific objective to destroy the actinide and long-life fission products from waste arising from the commercial nuclear programme. ATWs are considered in the molten salt thermal spectrum, see Figure 13.3, and liquid lead–bismuth in the fast spectrum.

13.8.4 CERN

CERN have put forward a conceptual design for a fast neutron operated high-power energy amplifier (EA) (Figure 13.4). The principles are described in detail in Carminati *et al*. (1993), Rubbia *et al*. (1994) and Andriamonje *et al*. (1995). More recent optimised realisations are described in Rubbia *et al*. (1997). The EA can operate for an indefinite period in a closed cycle. The fuel load is discharged, apart from fission fragments, and then re-introduced into the sub-critical unit, but with natural thorium introduced to compensate for burnt fuel. Equilibrium is achieved between burning and incineration after several cycles. This represents an extremely efficient use of fuel.

The EA module includes a 1500 MWt unit with a 1.0 GeV proton accelerator of 12.5 mA. The accelerator is a modular cyclotron; i.e. a plant is made up of a number of modules.

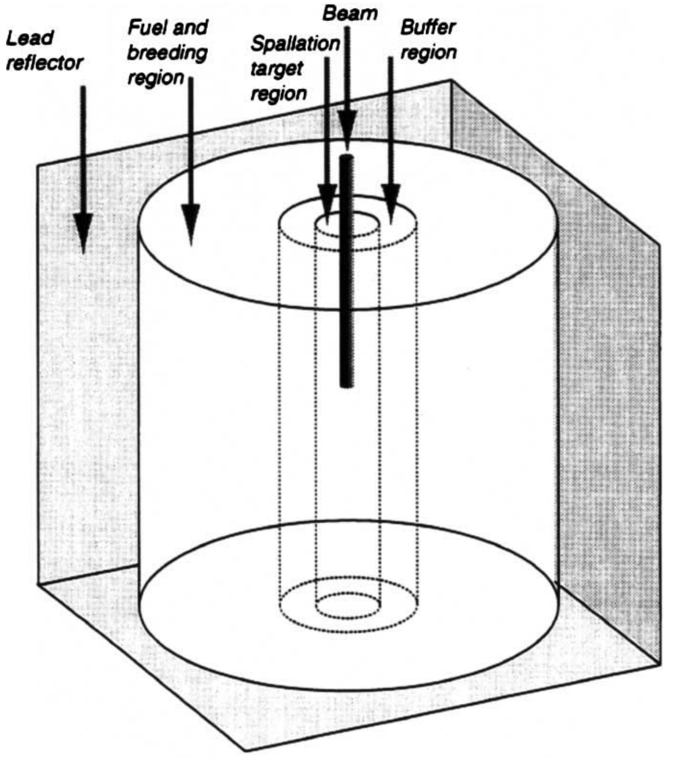

Figure 13.4. Conceptual design concept of the diffuser driven energy amplifier. Source: Carminati *et al*. (1993).

A fast neutron EA is envisaged if the EA has power commensurate with the current of large pressurised water reactors. The proton beam is a novel element of the design; the current is lower by one order of magnitude than most LINAC designs. The anticipated efficiency, i.e. the beam power over the mains load, is of the order of 40%. The beam penetrates the EA through an evacuated tube and tungsten window, specially designed to withstand radiation damage and thermal stress. The electrical energy to operate the accelerator is about 5% of the primary energy production.

The coolant is molten natural lead at a temperature of 600–700°C; lead being chosen because of its high boiling point (1743°C), which combined with the negative void coefficient of the EA, enables very high operating temperatures to be reached. Heat can be removed by natural convection.

The EA can operate with various different fuels, for plutonium to be transformed into ^{233}U, the EA would be initially loaded with actinide waste and thorium. Other actinides, e.g. americium, or neptunium could also be added. The EA mixture is sub-critical with k_{eff} in the range 0.96–0.98.

13.8.5 ITEP

The ITEP, Moscow, together with a number of other Institutes (Shvedov *et al*., 1994; Chuvillo and Kiselev, 1997), have conducted investigations in the use of ADS.

The main objectives are as previously discussed including a means of utilising large amounts of weapons-grade and commercial plutonium, and for waste management, etc. These developments are being considered against possible future scenarios for the Russian nuclear power industry, e.g. continuing the development of new generation NPPs, which would include improved VVER type reactors and possibly BN-800 designed fast reactors.

Different modes of operation are being considered including transmutation with and without power utilisation, the production of new fissionable materials and long lived radioactive waste transmutation, and the utilisation of NPP spent fuel assemblies as nuclear fuel. Within these modes, fuel cycles include uranium, plutonium, uranium–thorium, plutonium–thorium and other actinide fuel cycles.

Both solid and liquid fuels are being considered. The use of oxide fuel and zirconium cladding would enable advantage to be taken of existing fuel production experience. MOX fuels with plutonium, cermet and nitride fuels and actinide addition into MOX fuel may be future options.

ITEP have been investigating fluoride molten salts of the type $Li{-}BeF_2{-}ThF_4{-}Pu_4$, which have some advantages in reduced radiation damage, and reduction in the amount of fission products. However, the existing knowledge base on the performance of such fuels is more limited.

Different designs of blanket are being considered (IAEA-TECDOC-985, 1997b). These include blankets of solid fuel, blankets for liquid fuel and a modular channel blanket and a design for liquid fuel with a homogeneous blanket.

Conceptual targets include solid tungsten and other materials for proton currents up to 30 mA and liquid targets made of lead and lead–bismuth eutectic for high values of proton current. Figure 13.5 shows an example of a design with a lead–bismuth target with both fast and thermal blankets of Pu and Th oxides.

Experimental studies are being carried out at ITEP to verify the various concepts. These relate to both ADS design and to the selection of appropriate materials, e.g. in relation to the target and blanket.

13.8.6 CEA

CEA are conducting a research programme (Viala and Salvatores, 1994) on the potential of thermal or fast reactors for transmutation of waste in partnership with EdF, FRAMATOME and COGEMA (Salvatores *et al*., 1997a). Different laboratories within CEA are working within the ISAAC programme on the physics of ADS including accelerator technology, the physics of source driven multiplying systems and spallation physics.

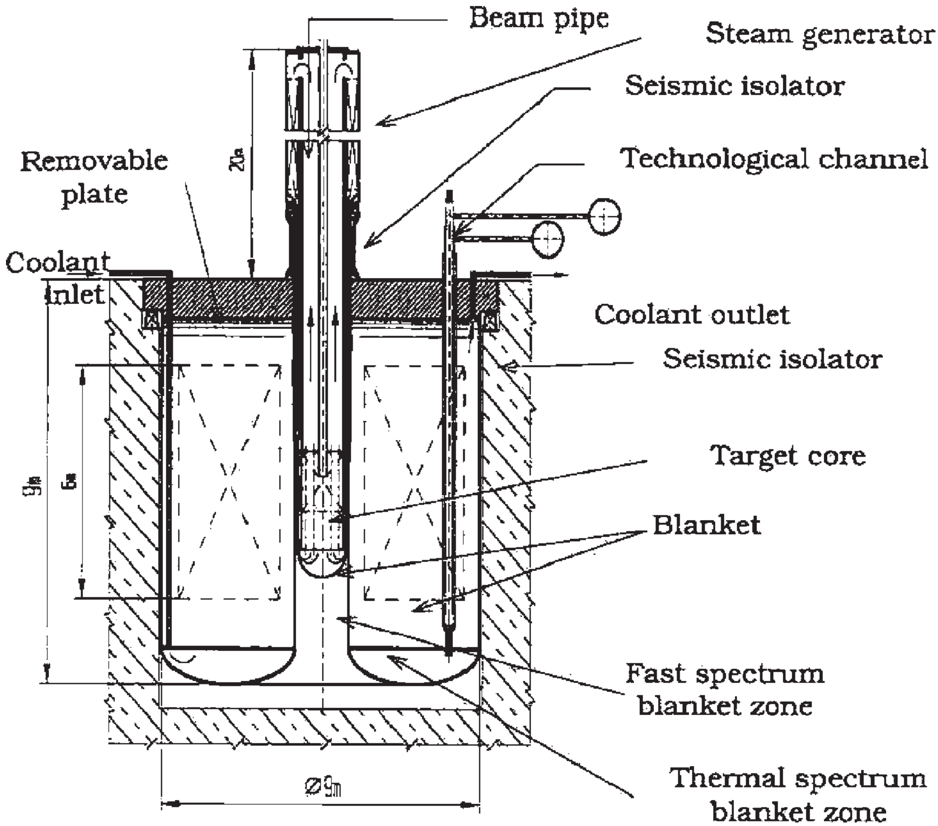

Figure 13.5. Lead–bismuth target and blankets. Source: Shvedov *et al*. (1997).

Feasibility work on accelerator structures with coupled cells and on beam dynamics has been carried out. Theoretical studies on high intensity accelerators have been carried out in support of experiments (FODO, on the beam dynamics) and (SATURNE, on the design of a 100 mA proton source) (IAEA-TECDOC-985, 1997a).

The physics of multiple sub-critical systems plays a central role and are being studied in several experimental programmes. The MUSE experiments (Salvatores *et al*., 1997b) in the MASURCA facility in CADARACHE are providing understanding on the neutron source and the impact of the source spectrum and environment at different levels of sub-criticality. Supporting experiments to determine actinide and fission product cross-sections are being carried out in the Geel LINAC, other experiments were also carried out in Superphénix.

On the spallation physics, thin and thick targets, spallation residual nuclei measurements, differential cross-sections' measurements and neutron production rates are being studied in the SATURNE experimental programme. Codes to model cascades include the code system SPARTE, supported by the Monte Carlo code TRIPOLI and the nuclide time evolution code DAEWIN. Future work envisages the coupling with the standard neutronics code ERANOS (Doriath *et al*., 1994).

System studies have been performed based on various scenarios, in which an ADS is used to develop a relatively clean source of nuclear energy within a fuel cycle, where LLFP are eliminated and radioactive wastes are concentrated in a small number of facilities in a nuclear reactor park.

Basic nuclear and particle physics is performed by the Institute National de Physique, et de Physique des Particles of CNRS, and also in the Direction Des Sciences de la Materie of CEA. The PRACEN research programme was set up in these laboratories to perform radiochemical studies within nuclear storage facilities. A recent joint research programme 'GEDEON', involving a collaboration between CNRS, EDF and CEA, has been set up to encompass the common areas of interest of the ISAAC and PRACEN programmes and to explore innovative options for waste management.

13.9. SAFETY

13.9.1 Scenarios

It can be seen from the earlier discussion that there are different ADS concepts being considered based on a number on different sub-critical reactor types.

Some of these reactor types have attracted considerable levels of safety research in regard to critical reactor operation. In principle, there are similar categories of accidents that could occur in ADS sub-critical reactors as could occur in critical reactors (Wider, 1997), see, e.g. Table 13.5.

ADS discussed above have included fast systems with solid fuel and liquid metal (lead or sodium) cooling and fast systems with circulating molten salt/MA. Fast reactors with

Table 13.5. Safety analysis

Reactor system	Event	Safety function status
Low pressure/fast and thermal	LOF, LOHS, LRHR	
	Additionally	Accelerator beam switched off?
High pressure	LOCAs	
Fast/thermal systems	TOP/RIA	Accelerator beam not switched off?
All systems	Accelerator over-power	

Wider (1997).

gas cooling have also attracted some attention previously. Thermal systems have been considered with circulating molten salt/minor actinide/Pu and graphite moderator. Thermal systems have also been considered with molten salt/actinide/Pu or a water/oxide slurry circulating in pipes with heavy water moderator.

In low-pressure fast or thermal reactor systems, various categories of loss of cooling accidents can occur. Typical examples are loss of flow (LoF) due to pump failure, or loss of heat sink (LoHS) due to pump failure in the secondary heat removal loops, or feedwater pump failure. Loss of decay heat removal is another example.

For high-pressure systems, loss of coolant accidents (LoCAs) are an additional possibility, occurring due to a break or leak leading to a sudden depressurisation, e.g. in a gas-cooled fast reactor.

ADS could also be vulnerable to transient overpower (TOP) in the case of fast reactors. Concerns for fast systems include possible reactivity insertions associated with moderator insertion, or a possible positive void coefficient in the case of a sodium-cooled fast reactor. Under more extreme accident conditions and core meltdown, reactivity insertions could result from fuel movement. Reactivity induced accidents (RIAs) are a possible concern in thermal reactors. In all systems, inadvertent withdrawal of control rods or control rod ejection in pressurised systems are possible scenarios although since ADS are sub-critical there may be fewer control rods than in critical reactors. The accumulation of fissile material in circulating liquid fuel systems or due to extreme perturbations, e.g. due to earthquakes could cause reactivity insertion. Finally there is the question on whether there are scenarios leading to a sudden increase in accelerator power.

As for a conventional critical reactor, a high degree of reliability is required for the operation of the key safety functions. In the case of an ADS, the most important requirements are the accelerator shut-off system and the decay heat removal systems.

13.9.2 Switching Off the Accelerator Beam

If the accelerator beam is switched off, the external spallation source will turn off and the reactor will go sub-critical with the power at decay heat levels. In a critical reactor, shutdown is achieved via the mechanical insertion of control rods. In both cases there is a delay from the trip signals for shutting off the beam or for activating the control rod release, which might be of the order of 0.5 s. However, overall, the time to switch off the current to the accelerator would be much faster than the control insertion time in a critical reactor, e.g. 1.5–3 s in a PWR (a little faster for a fast reactor with a smaller core).

A number of different beam shut-off systems are being considered. Diverse trip signals are necessary that result in beam shut-off. Since shut-off is important in cooling failure accidents, the current could be coupled to that driving the coolant pumps on the various cooling loops or on the feedwater pumps. Other passive means involve dropping the spallation target. This could be achieved by supporting with a low melting point metallic structure, which would melt in the event of sufficient temperature increase. Another could be via a magnetic structure, which would drop the target once the Curie temperature is reached. Other methods include deflecting the proton beam or, as in the ADS Rubbia design, by interrupting the beam by the rising lead level in the event of a cooling failure.

13.9.3 Cooling Failure Accidents with Spallation Beam Still Working

13.9.3.1 Sodium-Cooled Fast ADS. ADS have more 'inertia' than the corresponding critical reactor in that they are less sensitive to both positive and negative feedbacks (Bell, 1994). For example, with the source still on, they have lower but wider power peaks than in the critical reactor. In the ADS, the power rises earlier due to the lesser influence of negative feedbacks such as Doppler, axial expansion and structural effects. It falls later due to the lesser influence of fuel dispersion. In the sodium voiding phase, pin failures could occur.

The more sub-critical the ADS, the more the above features are seen. To avoid core meltdown, the source must be switched off, before much sodium voiding occurs. As stated earlier, fuel slumping can lead to re-criticality and power excursion because in a fast system the core is not in its most critical configuration prior to the event (Theofanous and Bell, 1985).

The ADS has some advantages over the critical reactor in that the time constants for the power excursion are longer and rapid power excursions are not possible at least when the ADS is in its original configuration. There is similarly a longer time period to detect sodium boiling or pin failures and hence initiate a beam switch-off.

13.9.3.2 Gas-Cooled Fast ADS. Gas-cooled fast ADS share the similar advantages and disadvantages that gas-cooled fast critical reactors have compared with fast sodium systems. Advantages include the utilisation of a chemically inert gas, and it may be possible to use water for post-accident cooling, e.g. if an in- or ex-vessel core-catcher can be designed and concerns of re-criticalities can be addressed.

Gas-cooled systems can also clearly suffer cooling failure events, but since system pressures are comparatively much higher than the sodium coolant system, LOCA accidents are an additional issue. In all cases, shut-off of the beam is crucial for preventing a core melt. A disadvantage of the gas system is that decay heat removal cannot be achieved solely by natural convection, thus back-up diesel generators are needed to be on stand-by in the event of loss of power to the active circulation pumps.

13.9.3.3 Lead-Cooled Fast ADS. Lead coolant has a number of advantages as a coolant compared with liquid sodium. A lead system would not suffer from positive feedback effects on reactivity in the event of boiling. It is only a weak moderator and, therefore changes in reactivity do not result due to changes in density effects.

It is also relatively chemically inert to air and water. The one disadvantage is the relatively high melting point 327°C, which means electrical heating would be required during start-up and there may be the possibility of freezing and blockage in the event of electrical system failure.

The Rubbia design includes lead as a coolant and it has been analysed against cooling failure transients such as LOF due to pump failure and LOHS due to loss of feedwater. The system has good natural circulation cooling characteristics so LOF is not an issue. For LOHS, meltdown could occur if the beam is not shut-off. This could also occur for slow reactivity insertions under similar conditions. However, the Rubbia system incorporates a specific provision to shut off the beam, based on shielding of the target by a rising liquid lead level under accident conditions. Further provision is also included in the Rubbia design to ensure long-term removal of decay heat by air natural circulation of the guard vessel.

13.9.3.4 Thermal ADS with a Circulating Salt–Fuel Mixture. Thermal systems have generally larger cores than fast systems because power densities are lower. This has the advantage of greater thermal inertia under coolant failure accident conditions allowing more time for accident detection, prevention or mitigation. Without switch-off of the beam though, pressurisation, heat-up and boiling would occur. However, this could be mitigated by spallation target melting and material movement leading to neutronic shut-down.

The time-scale for decay heat-up of the larger core systems is of the order of tens of hours since the fuel is distributed around the core and primary circuit and the system is in natural circulation mode. Some long-term cooling system/procedure though would need to be established, i.e. the salt–fuel mixture may need to be drained into a cooled tank. For smaller systems it may be possible to remove all the heat via natural circulation.

An advantage of a liquid fuel system is that short-lived fission products can be removed to reduce the fission product inventory.

The precipitation of fuel or MA may be a concern in salt–fuel ADS. This phenomenon could lead to flow impedance and loss of cooling in selected areas but also the density variation around the circuit could lead to criticality concerns.

There is also some concern that loss of cooling could lead to power increases due to a positive temperature coefficient in pure salt/Pu/minor actinide mixtures, since no ^{238}U or ^{232}Th with their absorption resonances would be present.

There is also the issue of possible explosive contact between molten salt and water; there may be potential for this event in some designs (Hohmann *et al*., 1982).

Inspection of components is also difficult in molten salt–fuel systems because pumps and heat exchangers become contaminated with radioactive material. Furthermore leaks would result in contamination of the whole containment.

13.9.4 Reactivity Accidents with the Accelerator Beam On

13.9.4.1 TOP Accident in a Sodium-Cooled ADS. Since the sodium-cooled ADS is initially sub-critical, fast or medium ramp rates can be accommodated for sizeable reactivity insertions. For example, calculations (Wider, 1997) of \$170 per second for a total insertion of \$2.65 at a sub-criticality of −\$3 and \$6 per second for a total insertion of up to \$3 for a sub-criticality of −\$5 showed no significant power excursion. For a critical reactor, the power would be 2000 times the nominal power.

For slower ramp rates, e.g. at \$0.1\$ per second, for a similar total insertion of about \$3 at a sub-criticality of −\$3 per second, a failure of a single channel occurred. At a sub-criticality of −\$5, failure also occurred but later, and it did not occur at all for a −\$10 sub-criticality. This compares with the critical reactor case in which pin failures in 1 out of 10 channels occur.

Thus for fast ramp rates without scram, the ADS behaves in a benign manner, for slow ramp rates there may be limited core damage at a later stage for insufficient sub-criticality or failure to scram the beam.

13.9.4.2 TOP and RIA Accidents in a Gas-Cooled ADS. Regarding fast and medium ramps, the gas-cooled fast ADS would behave similarly to the sodium-cooled ADS, i.e. benignly. For slower ramp rates, there could be rather more pin failures, because fuel dispersal may be less and, therefore provide less negative feedback.

13.9.4.3 TOP and RIA Accidents in a Thermal Molten Salt–Fuel Mixture. A thermal ADS would also act benignly under fast or medium ramp insertions. This system would also probably show an advantage for slower insertions compared with the fast systems above, because with a fluid system, the pins would probably not fail.

13.10. FUTURE ACTIVITIES

It is clear that ADS offer some interesting additional features that complement the conventional critical reactor technologies that currently exist or that may be considered in the future. However, there would need to be significant investment, research and

Table 13.6. Future activities

Technical research in selected fields
Investigation of different fuel cycles and energy systems for different applications
Further experimental programmes in demonstrating ADS feasibility
Extension of nuclear data into the ADS applications
Increased international collaboration and information exchange

IAEA-TECDOC-985 (1997a).

development in ADS technology if these systems are to become available commercially. Future activities that could be foreseen are outlined in Table 13.6. Clearly ADS would be subject to the same economic competitive pressures that face existing nuclear plant. Similarly ADS would have to meet the increasingly more stringent safety and environmental standards that are being imposed for licensing.

REFERENCES

Andriamonje, S., Angelopoulos, A., Apostolakis, A., Attale, F., Brillard, L., Buono, S., Calero, J., Carminati, F., Casagrande, F., Cennini, P. *et al.* (1995) Experimental determination of the energy generated in nuclear cascades by a high energy beam. *Phys. Lett. B*, **348**, 697–709.

Bell, C.R. (1994) Safety features of sub-critical fluid fuelled systems, International Conference on Accelerator-Driven Transmutation Technologies and Applications, Las Vegas, US.

Bonnaue, P., Rief, H., Mandrillon, P. & Takahashi, H. (1986) Actinide transmutation by spallation in the light of recent cyclotron development, NEACRP-A-910, Session B. 1.2, 1987. (European/American) Reactor Physics Committee Report and ICENES 86, Madrid, July 1986.

Bowman, C.D. (1997) Basis and objectives of the Los Alamos driven transmutation technology product, Status Report, IAEA-TECDOC-985, November 1997.

Carminati, F., Geles, C., Klapisch, R., Revol, J.P., Roche, Ch., Rubio, J.A. & Rubbia, C. (1993) An energy amplifier for cleaner and inexhaustible nuclear energy production driven by a particle beam accelerator, CERN/AT/93-47 (ET).

Carminati, F. *et al*. (1994) An energy amplifier for cleaner and inexhaustable nuclear energy production driven by a particle beam accelerator, CERN/AT/93-47 (ET).

Chuvillo, I. & Kiselev, G. (1997) ITEP concept of the use of electro-nuclear facilities in the atomic power industry, Status Report, IAEA-TECDOC-985, November 1997.

Coleman, W.A. & Armstrong, T.W. (1970) NMTC Monte Carlo Nucleon Meson Transport Code System, ORNL 4606.

Cowell, B.S., Krakowski, R.A. *et al*. (1995) Accelerator-based conversion (ABC) of weapons plutonium: plant layout study and related design issues, Los Alamos National Laboratory Report, LA-UR-95-1096.

Doriath, J.Y., Ruggieri, J.M., Buzzi, G., Rieunier, J.M., Smith, P., Fougeras, P. & Maghnouj, A. (1994) Reactor analysis using a variational nodal method implemented in the ERANOS System, ANS Topical Meeting, Knoxville, TN, US, vol. II, p. 20.

Hohmann, H., Kottowski, H., Schins, H. & Henry, R.E. (1982) Experimental investigations of spontaneous and triggered vapour explosions in the molten salt/water system, International Meeting on Thermal Nuclear Safety, Chicago, US.

IAEA-TECDOC-985 (1997a) Accelerator driven systems – energy generation and transmutation of nuclear waste, Status Report, IAEA-TECDOC-985, November 1997.

IAEA-TECDOC-985 (1997b) Physical features and performance accelerator driven systems (ADSs), Status Report, IAEA-TECDOC-985, November 1997.

Nakahara, Y. & Tsutsui, T. (1982) NMTC/JAERI – a simulation code system for high energy nuclear reactions and nucleon-meson transport process, JAERI Memorandum M82-198.

Nishida, T. *et al*. (1990) Improvement of spallation reaction simulation codes NMTC/JAERI and NUCLEUS, Second International Symposium on Advanced Nuclear Energy Research, Mito.

Prael, R. & Lichtenatein, H. (1989) LA-UR-89-3014.

Radiation Shielding Information Centre (1977) HETC Monte Carlo high-energy nucleon meson transport code, Report CCC-178, ORNL.

Ranft, J. *et al*. (1985) FLUKA82, CERN Divisional Report TIS-RP/156/CF.

Rubbia, C. *et al*. (1994) A high gain energy amplifier operated with fast neutrons, Las Vegas Conference on Accelerator Driven Transmutation Technologies and Applications, 25–29 July 1994.

Rubbia, C. *et al*. (1995) Conceptual design of a fast neutron operated high power energy amplifier, CERN/AT/95-44 (ET), European Organisation for Nuclear Research.

Rubbia, C., Rubio, J.A., Buono, S., Carminati, F., Fietier, N., Galvez, J., Geles, C., Kadi, Y., Klapisch, R., Mandrillon, P., Revol, J.P. & Roche, Ch. (1997) CERN Group conceptual design of a fast neutron operated high power amplifier, Status Report, IAEA-TECDOC-985, November 1997.

Salvatores, M., Slessarev, I. & Uematsu, M. (1995) Physics characteristics of nuclear power systems with reduced long-term radioactivity. *Nucl. Sci. Engng*, **120**, 18–39.

Salvatores, M., Martini, M., Slessarev, I., Soule, R., Cabrillat, J.C., Chauvin, J.P., Finck, Ph., Jacqmin, R. & Tchistiaakov, A. (1997a) MUSE-1: a first experiment at MASURCA to validate the physics of sub-critical multiplying systems relevant to ADS, Status Report, IAEA-TECDOC-985, November 1997.

Salvatores, M., Schapira, J.P. & Mouney, H. (1997b) ADS programmes in France: French programmes for advanced waste management options, Status Report, IAEA-TECDOC-985, November 1997.

Shvedov, O.V., Chuvilo, I.V., Kulikov, E.V., Vasiliev, V.V., Igumnov, M.M., Kozodaev, A.M., Volkov, E.B. & Lopatkin, A.V. (1994) ITEP sub-critical neutron generator driven by a charged particle accelerator, Report for the International Conference on Accelerator-Driven Transmutation Technologies and Application, Las Vegas, US, July 25–29, 1994.

Shvedov, O.V., Murin, B.P., Kochurov, B.P., Shubin, Yu.N., Volk, V.I. & Bogdanov, P.V. (1997) Weapons plutonium in accelerator driven power system, Status Report, IAEA-TECDOC-985, November 1997.

Slessarev, I. (1997) Accelerator driven systems (ADS), a principal neutronics and transmutation potential, Status Report, IAEA-TECDOC-985, November 1997.

Takada, H. *et al*. (1992) Production of radioactive nuclides in a lead assembly with 500 MeV protons, *Proceedings of Specialists Meeting on Accelerator Based Transmutation, PSI.*

Takahashi, H. (1984) Fission reaction in high energy cascade, BNL-NCS-51245.

Takahashi, H. (1990) The role of accelerator in the nuclear fuel cycle, *Proceedings of the Second International Nuclear Energy Research, Mito, JAERI, Jan 24–26, 1990.*

Takahashi, H. (1997) Brookhaven National Laboratory ADS Concepts (US), Status Report, IAEA-TECDOC-985, November 1997.

Takahashi, H. & Gudowski, W. (1997) Research and development – general issues, Status Report, IAEA-TECDOC-985, November 1997.

Takahashi, H., Takashita, H. & Chen, X. (1995) Transmutation of high-level radioactive waste and production of ^{233}U using an accelerator driven reactor, *AIP Conference Proceedings 346 on International Conference on Accelerator-Driven Transmutation Technologies and Applications, Las Vegas, NV, July 1994*, AIP.

Takizuka, T. (1997) JAERI accelerator driven system project, Status Report, IAEA-TECDOC-985, November 1997.

Takizuka, T. *et al*. (1992) Conceptual design study of an accelerator-based actinide transmutation plant with sodium-cooled solid target/core, *Proceedings of the International Information Exchange Meeting on Actinide and Fission Product Separation and Transmutation, ANL.*

Theofanous, T.G. & Bell, C.R. (1985) An assessment of CRBR core disruptive accident energetics, *Proceedings of the International Topical Meeting on Fast Reactor Safety*, *vol. 1, Knoxville, US*.

Van Tuyle, G., Todopso, M., Geiger, M., Aronson, A. & Takahashi, H. (1993) Accelerator driven sub-critical target concept for transmutation of nuclear wastes. *Nucl. Technol.*, **101**, 1.

VanGinnekin, A. & Borak, T. (1971) *IEEE Trans. Nucl. Sci.*, **NS-8** (3), 746.

Viala, M. & Salvatores, M. (1994) An overview of the SPIN programme, *Proceedings of Third International Information Exchange Meeting on Partitioning and Transmutation, Cadarache*, OECD-NEA/PT Report N. 13, December 1994.

Wechsler, M.S., Lin, C. & Sommer, W.F. (1995) Basic aspects of spallation radiation damage to materials, *AIP Conference Proceedings 346 on International Conference on Accelerator-Driven Transmutation Technologies and Applications, Las Vegas, NV, July 1994*, AIP.

Weinberg, A.M. (1970) Molten salt reactors. *Nucl. Appl. Technol.*, **8**, 102–219.

Wider, H.U. (1997) Basic safety features of ADS, Status Report, IAEA-TECDOC-985, November 1997.

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Chapter 14

Nuclear Heat and Other Applications

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Chapter 14

Nuclear Heat and Other Applications

14.1. INTRODUCTION/OBJECTIVES

The development of nuclear power has been primarily concerned with electricity generation. However, there is increasing interest in utilising nuclear power for other purposes. Some of these have already been described in the two preceding Chapters 12 and 13, including systems for the destruction of plutonium, the conversion of minor actinides in waste and for the production of hydrogen. This chapter covers more generally, further applications of nuclear plant for other than electricity generation, e.g. reactor systems for district heating, desalination and other process plant.

Nuclear energy can provide an alternative to carbon fuels as a useful heat source. This was realised early in the history of nuclear power development. Nuclear reactors have already been utilised in many of the nuclear operating countries for supplying energy for district heating, seawater desalination and other industrial processes. Much of this energy has been produced from power reactors operating in co-generation mode with electricity production together with one of the heat applications above.

IAEA (IAEA-TECDOC-1056, 1998) is acting as a forum to facilitate interest in nuclear heat applications. It has co-ordinated reviews of progress in the technology, including operating experience, technological developments and experience in the above applications. There are now over 60 reactors supplying heat in district heating, desalination and other industrial processes together with over 500 reactor-years of operational experience. The technical or safety-related issues in regard to nuclear heat applications have been considered in the international community. There are few additional issues compared with electricity generation applications.

Of the overall world energy consumption, about one third is used for electricity generation. Of the remainder, heat utilised by residential and industrial consumers represents a major share, the majority of this heat produced by burning fossil fuels, coal, gas, oil and wood. The next significant energy consumer is transport. Nuclear energy supplies about 6% of the world energy requirement and about 17% of the electrical supply. Although only about 1% of the heat produced by nuclear reactors is used for heat applications there are some signs of growing interest (Csik and Kupitz, 1997). Significant experience in co-generation of electricity and heat has been gained in Russia, Europe, North America and Japan, dedicated heat producing plants are now also receiving attention, e.g. Russia and China (IAEA-TECDOC-1056, 1998).

Historically, there has been more interest in district and process heat applications than in desalination. However, with the obvious requirements for freshwater in the developing countries, there is increasing interest in desalination applications in the IAEA Member States (IAEA, 1998). There is likely to be an increasing need for freshwater in much of the developing world over the next few decades (Wangnick, 1995).

High-temperature applications are again mentioned briefly in this chapter to complete the survey. There are some additional applications (other than hydrogen production) under consideration in some countries.

14.2. NUCLEAR HEAT APPLICATIONS

The temperature requirements for different heat applications vary considerably. Temperatures of the order of 100°C are required for district heating and seawater desalination whereas for some process heat applications and hydrogen production temperatures of the order of 1000°C and above are required. Different reactor types supply different temperature ranges of output, typical ranges are shown below in Table 14.1.

There is a wide range of applications, and different applications have different requirements, particularly temperature requirements (Table 14.2). The lower temperature end with water reactors and the higher end with high-temperature gas reactors have received the most attention to date (IAEA-TECDOC-923, 1997). A standard requirement for most users is reliability and availability. This is particularly so for the process industry, where production depends on energy supply to continue. In industry, energy must usually be available as a base-load commodity. This contrasts the load requirements for district heating where the demand is dependent on climatic conditions. Consequently load factors for energy producers for district heating applications may be much smaller than those required for industrial applications.

For reactors operating in co-generation mode for electricity and heat, there are issues of balance that need to be considered. For large power reactors, the main output may be electricity and these reactors will be optimised for base-load electricity generation.

Table 14.1. Temperature ranges available from different reactor types

Reactor type	Maximum temperature (°C)
Nuclear heating reactor (NHR)	200
Light water reactor (LWR)	320
Liquid metal reactor (LMR)	550
Advanced gas reactor (AGR)	650
High-temperature gas reactor (HTGR)	900
Very high temperature reactor (VHTR)	1500

IAEA-TECDOC-1056 (1998).

Table 14.2. Temperature requirements for different applications

Application	Temperature range (°C)
District heating	100–200
Desalination	100–200
Oil refining/processing of oil shale	250–600
Refinement of coal	400–950
Production of hydrogen	900–1000
Iron, cement, glass production	1000–1600

IAEA-TECDOC-1056 (1998).

For small reactors, a higher proportion of their output may be heat; therefore, significant fluctuation of the heat demand could result in fluctuation of electricity output. Thus the technology needs to ensure that the electricity production and the grid load are compatible.

In this chapter, some of the various reactor designs that are being considered for heat and other applications are discussed. To date, most of the operational experience has been on water-reactor systems. Some operational experience has been gained with liquid sodium. Lead and particularly lead–bismuth systems are being examined in Russia, both for district heating and for seawater desalination. As discussed in Chapters 12 and 13, various innovative reactor systems are being considered for high-temperature applications.

14.3. DISTRICT HEATING

District heating plants supplying hot water and steam are widely used in countries with cold winters such as Denmark, Finland, Sweden and Russia. Large cities require 600–1200 MWt, smaller communities perhaps 10–50 MWt (IAEA-TECDOC-1056, 1998). The heat is produced by extracting steam from low-pressure turbines (for base load) and/or high-pressure turbines (for peak heat demand) and then distributed in insulated pipelines. These are on the order of 10 km, the shortest being a few kilometers, the longest built is in excess of 20 km.

The majority of nuclear applications for district heating have been reactors operating in co-generation with electricity producing mode. Such plants have been operated in Bulgaria (Kozloduy), Germany (Greifswald), Hungary (Paks), Russia (Bilibino, Belojarsk, Balakovo, Kalinin, Kola, Kursk, and Sankt Petersburg), Slovakia (Bohunice), Switzerland (Beznau) and Ukraine (Rovno, South Ukraine).

With regard to dedicated heating reactors, there have been demonstration plants constructed and tested in Canada (SLOWPOKE) and also China (NHR-5). There has also been a research reactor operating in Russia (Obninsk) for more than 20 years.

Table 14.3. District heating water reactors

Reactor	Type	Rating (MWt)	Country
RUTA	LWR pool type	10–55	Russia
NHR	LWR pool type	5–200	China
KLT-40C	PWR	80 per unit	Russia
VK-50/300	BWR	50–300	Russia

Data from IAEA-TECDOC-1056 (1998), Adamov *et al*. (1995) and IEA/OECD (NEA)/IAEA (2002).

The plants include barriers to prevent any release of radioactivity into the grid network. A leak tight intermediate loop is added which operates at a pressure greater than that of the steam pressure taken from the turbine cycle. The loops are also subject to continuous monitoring. In about 500 reactor-years of operating heat supplying reactors, no radioactive contamination of the network has been reported.

While much experience exists with co-generation, small- and medium-sized reactors may be more appropriate for district and process heating and also desalination applications. These have been reviewed in IAEA-TECDOC-881 (1996).

Some of the newer concepts of reactors for district heating are shown in Table 14.3 and summarised below.

14.3.1 RUTA

The State Research Centre of the Russian Institute of Physics and Power Engineering (SRC RF-IPPE) is examining the potential of a series of RUTA pool type reactors from 10–50 MWt (Baranaev *et al*., 1998) (Figure 14.1). This builds on experience gained from the 10 MWt water–graphite reactor AM that has operated for district heating since 1976. The system is being developed to provide a flexible power source for supplying heat either to cities or to much smaller communities. There is a particular need in the more remote regions of Russia (Adamov and Romenkov, 1996).

The RUTA system (Adamov *et al*., 1995) is based on a single reactor vessel operating at low pressure and in general the reactor process parameters are low, e.g. no water boiling in the pool. The power density is also low at approximately 15 kW l^{-1}. The system is integral with the primary heat exchangers accommodated within the pool. Water circulates by natural circulation under both normal and accident conditions.

14.3.2 NHR

The Institute of Nuclear Energy Technology (INET) in China designed and has operated a low-temperature heating plant from the early 1980s (Dazhong *et al*., 1996; Zheng, 1998). This pool-type plant provided district heating to nearby buildings. Following this activity, a 5-MW thermal experimental vessel type reactor, NHR-5 was started in 1986 at INET and went into operation in 1989.

The NHR-5 has a number of new and innovative features, including natural circulation, passive safety systems, integrated geometry and hydraulic control rod drive systems. To ensure protection of users from radioactive contamination, an intermediate circuit was added. The operating pressure of the intermediate circuit is higher than the primary side to prevent any radioactivity release to the network.

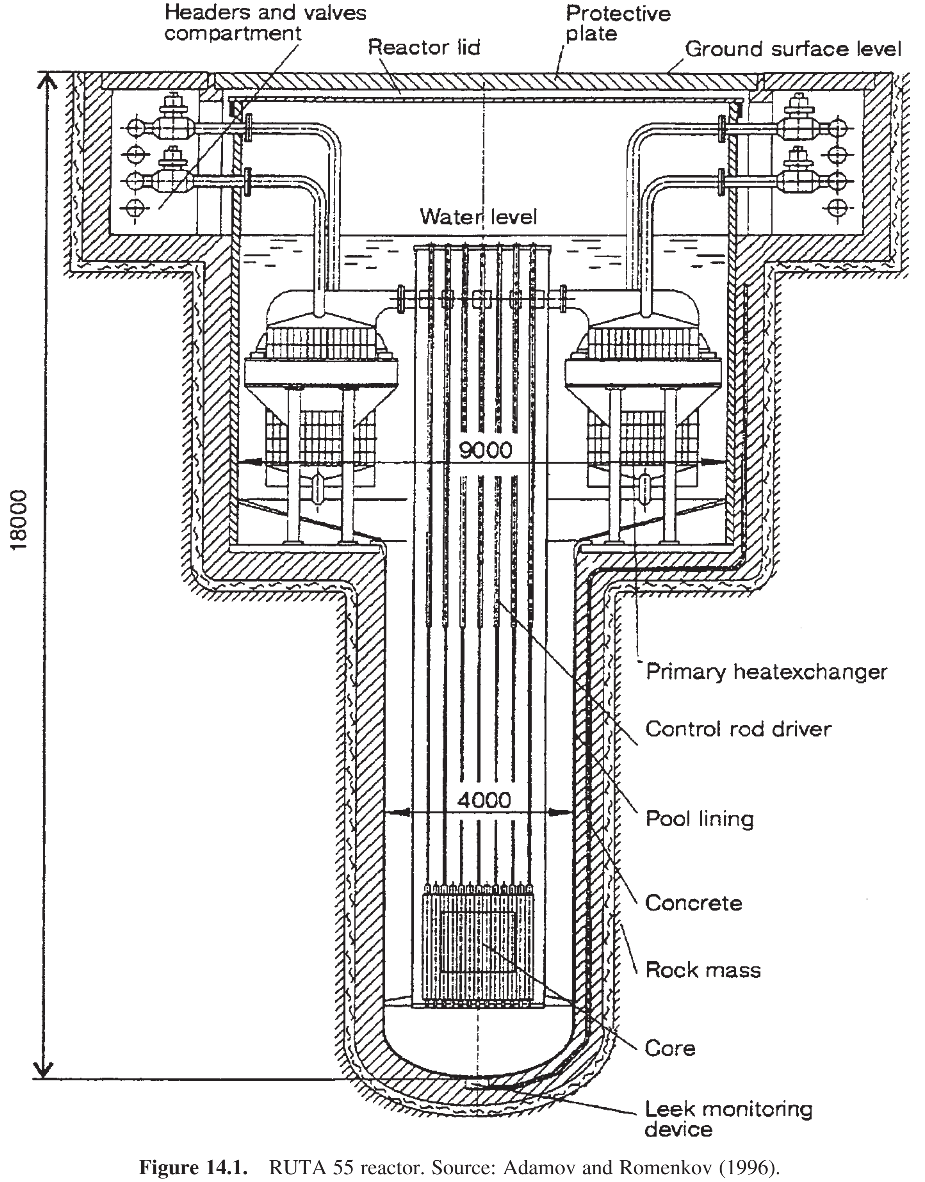


Figure 14.1. RUTA 55 reactor. Source: Adamov and Romenkov (1996).

A commercial scale NHR (NHR-200) with a generating capacity of 200 MWt has been developed since 1990, taking full advantage of the technology developed for NHR-5. Approval was given for building in 1995 at Daqing in China. The intention is that this technology can be applied to district heating, air conditioning, seawater desalination and other industrial processes.

14.3.3 KLT-40C

For many years Russia has developed marine nuclear reactors to power their ice-breaker transport fleet. Taking advantage of this experience, there are recommendations for KLT-40 type nuclear energy floating complexes to supply electricity and heat to remote regions in the far north and east of Russia.

This reactor type is being developed by the Experimental Machine Building Bureau (OKBM) of the Russian Federation. It incorporates the nuclear reactor in a barge and can, therefore, be regarded as a ship reactor (IEA/OECD (NEA)/IAEA, 2002). The floating power unit (FPU) is assembled in a factory. Factory fabrication can be optimised by the delivery of two units, including the steam side plant.

The plant is a conventional pressure-vessel loop type reactor including hydraulic loops, pumps, steam generator and pressure vessel. It contains a number of safety features including diverse and redundant shut-down and passive decay heat removal systems. There is self-regulation of the power levels at all power levels because of negative temperature and void coefficients.

There are reduced volumes of low and medium level waste. All the waste obtained over the 12-year operating cycle is stored on the FPU. The outlet temperatures are similar to other current PWRs.

14.3.4 VK-300

This is boiling water reactor concept considered for district heating and electricity supply developed by Research and Development Institute of Power Engineering (RDIPE), Russia (Zuznetsov *et al.*, 1998). It builds on the successful operation of the boiling water reactor VK-50, located in Dimitrovgrad, Russia. Outlet steam temperatures are around 285°C and the plant pressure is 7 MPa. The plant operates via natural circulation. Reactor neutronics provide negative feedbacks between reactor reactivity, its power, fuel temperature and quality. These reactors include relatively simple features, passive cooling and small primary pressure units and are, therefore, suitable for underground operation.

14.4. DESALINATION

There have been fewer application for desalination than for district heating. As for the latter case, the majority of applications have been with plants operating in co-generation mode, i.e. electricity and desalination. Desalination plants have been operated in Japan (Ikata, Ohi, Genkai, Takahama, Kashiwazaki). A range of different desalination processes have been used. There has also been some experience from a plant operating in the USA at the Diablo Canyon.

Other experience has been gained in Kazakhstan (Aktau) where the liquid metal cooled fast reactor BN-350 has been operating as a multi-energy source for electricity, drinking water and heat.

A non-nuclear facility was built in Israel for testing the nuclear desalination process. The heat source was produced by an oil-fired power plant N.B. this operated for only a short period.

Desalination is the process of obtaining freshwater suitable for drinking or industrial processes through the removal of salt from saline, usually seawater. This can be achieved using either distillation processes or via membrane processes using osmosis (IAEA-TECDOC-1056, 1998). Desalination processes include:

– Multi-stage-flash (MSF) distillation;
– Multiple-effect distillation (MSD);
– Reverse osmosis (RO).

Typical energy requirements and energy consumption rates for the three processes are shown in Table 14.4. These can be compared with the theoretical minimum energy requirement of 0.73 kW h m^{-3} for 35,000 ppm saline water at 25°C. The discrepancies are due to significant thermal processes and irreversibility that occur during the separation process.

14.4.1 Distillation Processes

In these processes, low-temperature steam is taken from the power plant turbine of the supplying plant to heat the saline solution. In commercial distillation, there are a number of heat recovery stages in series, because of the high heat of evaporation of water. These stages are at progressively lower pressures, resulting in flashing and mechanical vapour compression to occur.

In general, the more stages in place, the more efficient is the process. The number of stages is limited by both economic and technical reasons, e.g. the overall temperature

Table 14.4. Nuclear desalination energy requirements

Process	Heat consumption (kWt h m^{-3})	Electricity consumption (kWe h m^{-3})	Maximum brine temperature (°C)
MSF	45–120	3–6	120°C (brine recycle) 135°C (once-through)
MED	30–120	1.5–2.5	70°C (horizontal tube)

IAEA-TECDOC-1056 (1998).

difference between the heat source and the cooling water sink. The typical temperature reduction per stage for a commercial plant is 2–5°C. In terms of thermodynamic efficiency, expressed as kg of water produced against kg of steam used, the figure is 6–10 for MSF applications and up to 20 for MSD. These processes are described below.

14.4.1.1 MSF Distillation. In this process, seawater is passed through a number of stages where it is progressively heated (see below) until it reaches the main heating section supplied by the process heat source, see for example (IAEA-TECDOC-1056, 1998). The brine is then returned through these stages and freshwater is eventually obtained through a series of flashing and condensation processes. In particular, as the heated brine returning from the heat source passes into the first stage heat recovery section, flashing occurs due to pressure reduction. Vapour is produced which condenses on the entry pipe-work to the heating section within the first stage (providing the progressive heating referred to above). The condensate is collected in trays. This condensate together with the remaining brine (that has not flashed) is passed on the second stage. The process is then repeated for a number of stages and the separation process completed. Non-condensable gases are removed by a steam-jet ejector system. The seawater is also chemically treated to remove scale.

14.4.1.2 MED. This process also consists of a number of heat-exchange sections. At the first stage steam from the heating boiler passes through a tube bundle which is cooled by evaporating the entry seawater on the other side of the tube bundle. The resulting steam is then passed to a second stage heat exchanger. Any seawater not evaporated at the first stage is passed on to the second stage. The process is then repeated to complete the separation process. MED plants require similar scale removing processes as do MSF plants.

Several designs have been used. The main difference is in the design of the heat exchangers. The low-temperature horizontal tube multi-effect process (LT-HTME) has horizontal tubes and the brine is sprayed over the outside of the tubes. In the vertical-tube evaporation process (VTE), the evaporation is inside vertical tubes. The LT-HTME is the more dominant process used.

In general, MED plants are more efficient than MSF plants because their heat transfer processes are more efficient for given heat transfer area and similar temperature difference between the heat source and cooling water.

14.4.1.3 RO. RO is also used as a separation process (IAEA-TECDOC-1056, 1998). This process has been applied commercially and can produce freshwater down to between 100 and 200 ppm of total dissolved solids. The electricity consumption is in the range 4–7 kWe h m^{-3}.

In this process, seawater (brine) and water are held in a vessel in two-solution compartments separated by a semi-permeable membrane. Pressure is applied to the compartment containing the brine, sufficient to overcome the natural osmotic pressure of the solution and the permeate pressure (NB this is negligible compared with the natural osmotic pressure). In these circumstances, water flows from the brine compartment to the water compartment, the brine become more concentrated and purified water is obtained in the water compartment.

As the seawater is fed into the brine compartment, it is compressed up to 70–80 bars, sufficient to overcome a natural osmosis pressure of the saline solution of about 60 bars. In practice, only a portion of this water flows through the membrane, the remainder is discharged. The flow through the membrane is proportional to the pressure gradient of the applied pressure less the solution osmotic pressure. The proportionality factor depends on a range of factors including the geometry (shape, area, thickness) and the chemical properties of the membrane, the pressure, concentration, pH and temperature. Membranes have been used of varying design, spiral-wound, hollow fibre, also tubular, plate and frame type, the former two designs being the most commonly used.

14.4.1.4 Hybrid Desalination. Hybrid desalination systems can be used to combine power generation, with MSF or MED, and RO processes. This combined capability can be utilised to advantage in different ways, depending on the size and type of energy source available and the water quality product requirements. There are economic and technical advantages of hybrid as compared to single process technology.

These include the utilisation of a common seawater intake, optimised feedwater temperature for the RO plant, taking cooling water from MSF or MED plant, blending of product waters, common water treatments and various other optimisations that can be made through common process requirements. Some of the different hybrid desalination systems are reviewed in (Awerbuch, 1997).

Some of the reactor concepts that are under consideration for desalination applications are shown in Table 14.5 and discussed below.

14.4.2 RUTA-TE

A number of small medium-size nuclear power plants have been developed by RDIPE in Russia for district heating which can also be used for seawater desalination.

RUTA-TE is a pool-type thermal reactor that can be used for the cogeneration of electricity and power. The RUTA concept has already been described earlier in the book and more details are given in (RDIPE, 1994; Grechko *et al.*, 1998). It can be used in conjunction with an RO process or together with a distillation process. The latter is expected to be the more economic with this power source.

14.4.3 NHR-200/Desalination Plant

This concept is being considered by the Institute of Nuclear Energy, Tsinghua University, Beijing, China (Zhang *et al.*, 1998). The NHR-200 is an integral light water reactor

Table 14.5. Desalination water reactors

Reactor	Type	Rating (MWt)	Country
RUTA-TE	LWR pool type	70	Russia
NHR-200	LWR pool type	200	China
KLT-40C	PWR	80	Russia
NIKA-120M/300	PWR	70–300	Russia
UNITHERM	PWR	15	Russia
SMART	PWR	330 (Cogeneration)	Korea
MAPS	PHWR	200 MWe (Cogeneration)	India

Data from IAEA-TECDOC-1056 (1998) and IEA/OECD (NEA)/IAEA (2002).

(introduced in the previous section). The proposed options for the desalination system are based on the steam generator and MED process for water production as a single process or with a steam generator and MED process for co-generation of water and electricity (Duo *et al*., 1995).

14.4.4 KLT-40C

The Russian KLT-40C reactor could function within a seawater desalination complex, as a floating installation. The characteristics of the KLT-40 reactor were described in the previous section. The KLT-40 reactor could be used for desalination either within a distillation desalination facility (Chernozoobov *et al*., 1995) or within an RO facility (Humphries and Davies, 1995). Further details are also given in (Panov *et al*., 1998).

14.4.5 NIKA-120M & 300

These Russian-designed integrated PWR type reactors (Achkasov *et al*., 1997; Grechko *et al*., 1998) are being considered as either ground-based or floating nuclear reactors. They can be used in distillation, RO or hybrid mode. NIKA-120M would comprise two units, each of 70 MWt, whereas NIKA-300 is envisaged as a single unit at 300 MWt. The former is less economic than the latter. It is anticipated that a hybrid system could produce freshwater at a cost commensurate with that of the best fossil fuel plants.

14.4.6 UNITHERM

The UNITHERM Russian reactor concept (Adamovich *et al*., 1997) is based on a transportable small-sized PWR concept that can be delivered to site in components. It would be appropriate for remote areas. It is designed for a life of 20 years with a single fuel loading for flexibility of location and with no requirement for a local cooling water supply. It comprises a dual loop system, the primary loop driving a turbine, the secondary system supplies steam or hot water for the intended process. The UNITHERM power plant was found (Grechko *et al*., 1998) to have a higher freshwater cost than fossil fuel plants but these plants could be found to be more economic than alternatives in remote access regions.

14.4.7 SMART/Desalination Demonstration Plant

KAERI in Korea are developing an integrated desalination system coupled with the system integrated modular advanced reactor (SMART) 330 MWt integral PWR system (Chang and Kim, 1998) (Figure 14.2). SMART is an evolutionary PWR incorporating passive safety features, simplified systems, cost-effective component fabrication and a load-follow operational capability. It is designed to be a co-generation system, which aims to produce 40,000 m^3 per day of potable water, the remaining energy to be converted to electricity. The MSF and the RO options are both under review and investigation in Korea. The fundamentals of the SMART reactor design are shown in Seo (1997). Details on the licensing programme are given in Kim and Chang (1997).

14.4.8 MAPS/Desalination Plant

There is a proposal to set up a nuclear desalination demonstration plant at the Madras atomic power station (MAPS) in India. It will have a capacity of 6300 m^3 per day and would be based on the MSF and seawater RO processes (Hanra and Misra, 1998). The power would be drawn from a 200 MWe PHWR operating in co-generation mode.

14.5. LOW-TEMPERATURE PROCESS HEAT APPLICATIONS

In addition to district heat and desalination applications there have been various activities involving low-temperature process heat systems. Desalination is one such application, considered in the previous section.

This section considers some additional applications. These generally relate to the utilisation of process heating steam available from electricity producing reactors. Examples of various different industrial applications are given below (IAEA-TEC-DOC-1056, 1998).

The Gösgen 970 MWe PWR in Switzerland provides process steam to a cardboard factory at an output of 54 MWt. The Stade 640 MWe PWR in Germany supplied process steam for a salt refinery; further excess energy was supplied for space heating of another oil fired station nearby and a tank storage facility.

In Canada, the Bruce nuclear power development (BNPD) is a large nuclear electricity and steam-generating complex. It includes eight CANDU units with a total power output of 7200 MWe. The four 848 MWe units of Bruce A in conjunction with Bruce bulk steam system (BBSS) supplied up to 5350 MWt of medium pressure steam for the heavy water plants (HWPs). Additional energy was also supplied to the Bruce Energy Centre (BEC) industrial park.

New low-temperature process heat applications are likely to be based on existing technologies, suitably optimised. In terms of new concepts, a novel process heating water application is under consideration within the Argentinean mining industry (Table 14.6).

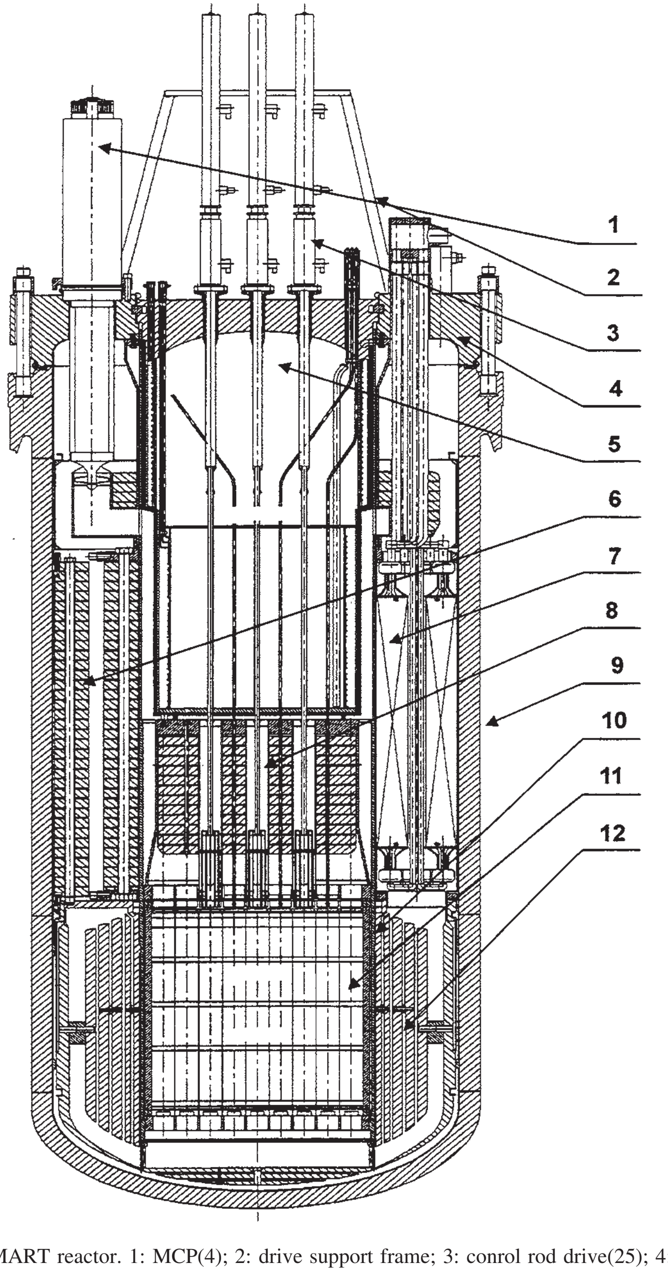


Figure 14.2. SMART reactor. 1: MCP(4); 2: drive support frame; 3: conrol rod drive(25); 4: annular cover; 5: pressuriser; 6: displacers; 7: steam generator(12); 8: shroud tubes; 9: reactor vessel; 10: core suppport barrel; 11: fuel assembly(57); 12: side screen. Source: Chang and Kim (1998).

Table 14.6. Process heating water reactors

Reactor	Type	Rating (MWt)	Country
CAREM-25	PWR	100	Argentina
MRX	PWR	Up to 300	Japan
KLT-40C	PWR	80 per unit	Russia
SMART	PWR	330 (Cogeneration)	Korea

Data from IAEA-TECDOC-1056 (1998) and IEA/OECD, NEA/IAEA (2002).

14.5.1 CAREM-25

CAREM-25 is an integral type PWR in which all components of the primary circuit (NSSS, pressuriser, primary heat exchangers and coolant pumps) are included in a single pressure vessel (IEA/OECD (NEA)/IAEA, 2002). It is being applied in Argentina as a dual purpose plant for electricity and process heat for the extraction and purification of various minerals including sodium sulphate.

A concern for smaller reactors of traditional design is that the economies of scale militate against the economics. With an integral design, savings can be made by reducing the number of pressure loaded and load-bearing components. Considerable emphasis is placed on inherent safety features in the design, including passive shutdown systems and decay heat rejection.

In addition to the economic competitive features mentioned above, there is scope for design modularisation and factory assembly. The other characteristics of operation relating to proliferation, waste management, resource efficiency, flexibility of operation (process heat and co-generation) are similar to those of other current generation PWRs of comparable size and application.

14.5.2 MRX

The MRX is a small PWR suitable for low-temperature process and cogeneration applications, developed by JAERI. It is based on a similar approach to CAREM-25 except that forced coolant circulation and all steam generators are required during full power operation.

Decay heat is dissipated by a passive heat removal system. There is a comprehensive containment system with a large volume of water available in the containment.

The remaining two systems KLT-40C and SMART in Table 14.6 have already been described in the previous two sections on district heating and desalination applications.

The temperatures available in MRX, KLT-40C and SMART for process heat applications are similar to that of current PWRs.

It is also worth mentioning that there are some additional low-temperature process heat applications for which nuclear heating could also be considered, e.g. urea synthesis and wood pulp processing (Institute of Nuclear Engineers, 2004).

Future developments in process heat applications are focussing on higher temperature reactor systems. These are considered in Section 14.8.

14.6. LOW-TEMPERATURE HEAT APPLICATIONS AND LIQUID METAL TECHNOLOGY

The utilisation of lead–bismuth reactors for district heating or for seawater desalination is being investigated in Russia (IAEA-TECDOC-1056, 1998). There are 150 reactor-years of experience of lead–bismuth reactor technology in Russia from application in the country's submarine fleet.

The technology is now being reassessed for either co-generating or single application district heating or desalination plants. Some of the plants, e.g. ANSTREM could also be used for refrigeration applications. The coolant has desirable chemical, activation and thermophysical properties, including low chemical reactivity with water, low long-lived induced gamma activity, negative void coefficient, a high boiling point and low freezing point. Some possible new designs for low temperature applications are shown in Table 14.7. The technologies under consideration include modular and small transportable plants such as ANSTREM with a compact reactor layout, but also larger plants such as BREST 300.

14.6.1 SVBR-75

The SVBR-75 reactor module is designed by EDB Gidropress and SSC RF IPPE for steam production to replace VVER-440 reactors that are being decommissioned (SSC RF-IPPE, EDB, 1996; Stepanov *et al.*, 1998) (Figure 14.3). Specifically it has been designed for application in the Novovoronezh power plant facilities as units 2, 3 and 4 are decommissioned. The concept is flexible and can be applied for combined generation of heat and electricity. The SVBR-75 concept exhibits the important features of lead–bismuth coolant systems (Gromov *et al.*, 1996).

Table 14.7. Liquid metal (lead–bismuth) reactors for heat applications

Reactor	Type	Rating (MWt)	Country
SVBR-75	LMR	250	Russia
ANSTREM	LMR	30	Russia
SC TNPTP	LMR	10	Russia
BREST 300	LMR	300 (MWe)	Russia
Energy Amplifier	LMR (sub-critical)	675 (MWe)	Europe

Data from IAEA-TECDOC-1056 (1998) and IEA/OECD, NEA/IAEA (2002).

14.6.2 ANSTREM

The ANGSTREM project (Stepanov *et al*., 1998) is based on the concept of a modular, transportable nuclear power and heating station, utilising fast reactor technology with lead–bismuth eutectic cooling. The main design organisation is EDB ‘Gidropress’ together with IPPE, Obninsk providing scientific consultancy. The ANGSTREM technology is envisaged for a number of applications including electricity generation, heat supply, freshwater and possibly hydrogen production.

14.6.3 SC TNPTP

A small capacity transportable nuclear power and technology plant (SC TNPTP) is being considered for electricity and heat supply, production of freshwater and also hydrogen (Komkova *et al*., 1998). The concept has been put forward by IPPE and St Petersburg Marine Building Bureau. The plant rating is chosen in order to optimise the economics for application of the reactor in remote areas in Russia. A 1-MWe unit prototype reactor TES-M has been designed but it is necessary to increase the power in SC TNPTP by at least a factor of 2, with no significant increase in the mass and dimensions to achieve satisfactory economics.

14.6.4 BREST 300

BREST 300 is a lead-cooled, pool-type fast reactor design operating at close to atmospheric pressure (IEA/OECD (NEA)/IAEA, 2002). The reference rating is 300 MWe. It has been put forward by RDIPE, Russia. It incorporates a loop concept for primary circuit heat removal. It is based on a relatively simple and robust design with passive decay heat removal to the environment. It has similar characteristics to those of the other lead-cooled reactors described above.

It has an increased core outlet temperature compared with the PWR making it a better candidate for somewhat higher temperature process heat applications.

14.6.5 Energy Amplifier

Lead-cooled subcritical reactors driven by a proton accelerator, such as the energy amplifier, are also being considered for process heat applications (IEA/OECD (NEA)/IAEA, 2002).

14.7. MEDIUM- AND HIGH-TEMPERATURE APPLICATIONS

The applications discussed previously in this section relate mainly to low-temperature applications. There are a number of interesting medium and higher temperature process

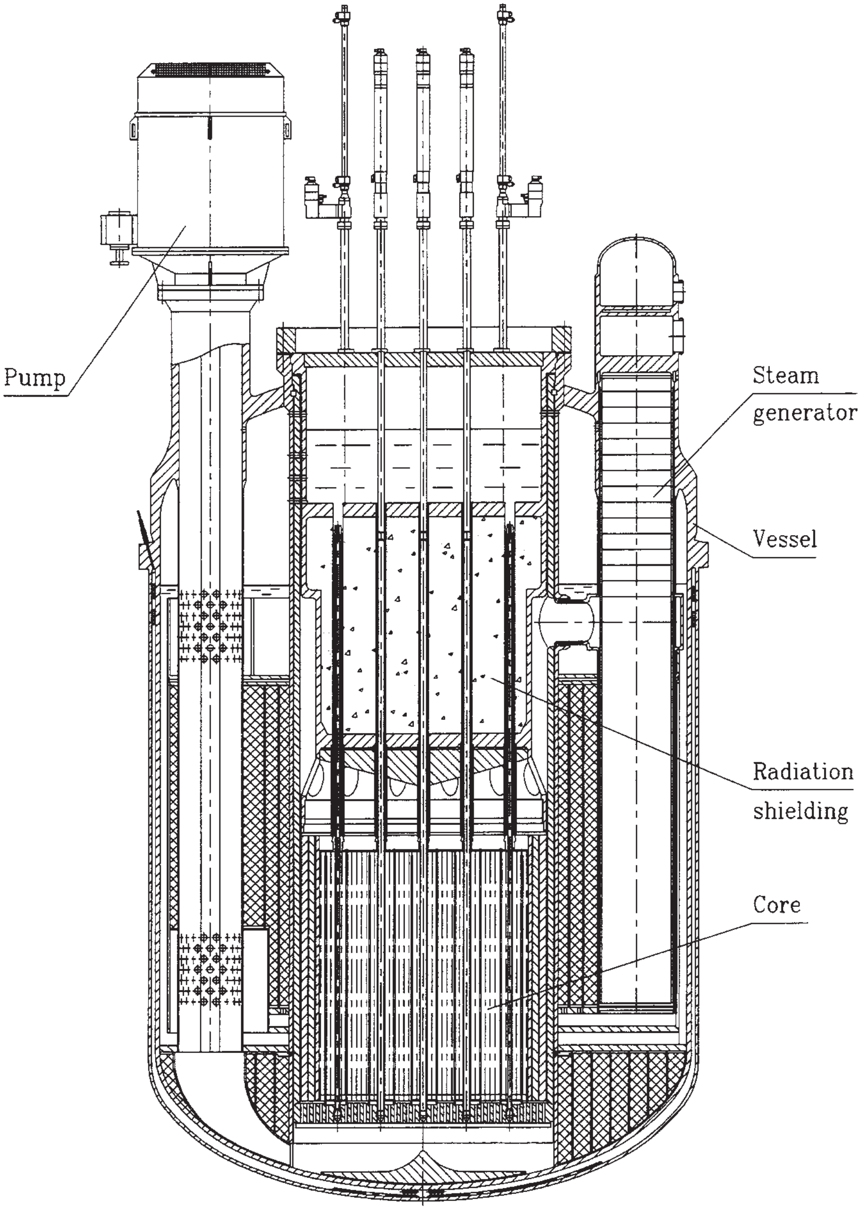


Figure 14.3. SVBR-75. Source: Stepanov *et al*. (1998).

heat-related applications associated with oil refining and liquid fuel production from coal and hydrogen production. A particular interest is in hydrogen production for future generation reactors; this has already been discussed in Chapter 12.

Nuclear heat applications at medium and high temperature have not yet been developed at industrial scale. They are, however, being researched at smaller laboratory scale. The most promising systems are the high-temperature gas cooled reactors. The present near-term designs (GT-MHR, PBMR) are evolutions from smaller prototype reactors that operated in the UK (Dragon), Germany (AVR & THTR-300), and the US (Peach Bottom and Fort St Vrain).

Some particular reactor concepts that are being considered in the development of future high- and medium-temperature applications are summarised in Table 14.8. These relate to on-going programmes in China, Japan and Russia. These reactor systems are described briefly below.

14.8. OIL REFINING

The utilisation of nuclear heating for application to various oil refinery processes is being considered in Russia. Thermal power is required at different medium-range temperatures for different operations. These include low-temperature processes up to 400°C associated with initial reprocessing of oil products, e.g. hydrocracking, hydrocleaning. There are also middle-range temperature processes up to 600°C associated with secondary oil refining processes, reforming, cracking, etc. The VGM-P, HTR-10 and BN-600 systems, described below, are seen as possible candidates for this application.

14.9. COAL REFINING

As noted in the previous section, there is continued interest in synthetic liquid fuel (SLF) production in China and Russia. The energy consumption and production in China is

Table 14.8. Medium- and high-temperature applications

Reactor	Type	Rating (MWt)	Country
VGM-P	Pebble-bed HTR	215	Russia
HTR-10	Pebble-bed HTR	10	China
BN-600/800	LMR	600/800 (MWe)	Russia
HTTR	Prismatic HTR	30	Japan
Generation IV	Various	Various	GIF countries

Data from IAEA-TECDOC-1056 (1998).

dominated by coal and there is a shortage of liquid fuel supply. The application of high-temperature reactors to convert coal to liquid fuel is, therefore, of interest.

There is also interest in Russia in converting low-grade brown coal to motor fuel. The VGM-P, HTR-10 and BN-600 systems, as described in the previous section, are seen as possible candidates for refining of coal, which requires still higher temperatures than are needed for oil refinement. The high-temperature gas reactors could also be used in the production of hydrogen, ammonia and mineral fertilisers by, e.g. methane steam conversion or other means.

14.9.1 VGM-P

The VGM-P is a pilot industrial modular helium-cooled reactor designed by OKB Mechanical Engineering, Russia (Figure 14.4). It is based on a pebble bed design approach that can be fuelled on line. It is being considered as a heat source for the various applications above (Golovko *et al*., 1995, 1998).

Oil and coal refining require different temperature levels. The industry requires high-temperature capabilities for the production of diesel from coal and for the production of hydrogen and fertilisers, etc., intermediate temperatures for secondary reprocessing of oil products and cracking, and low temperatures for initial reprocessing of oil products.

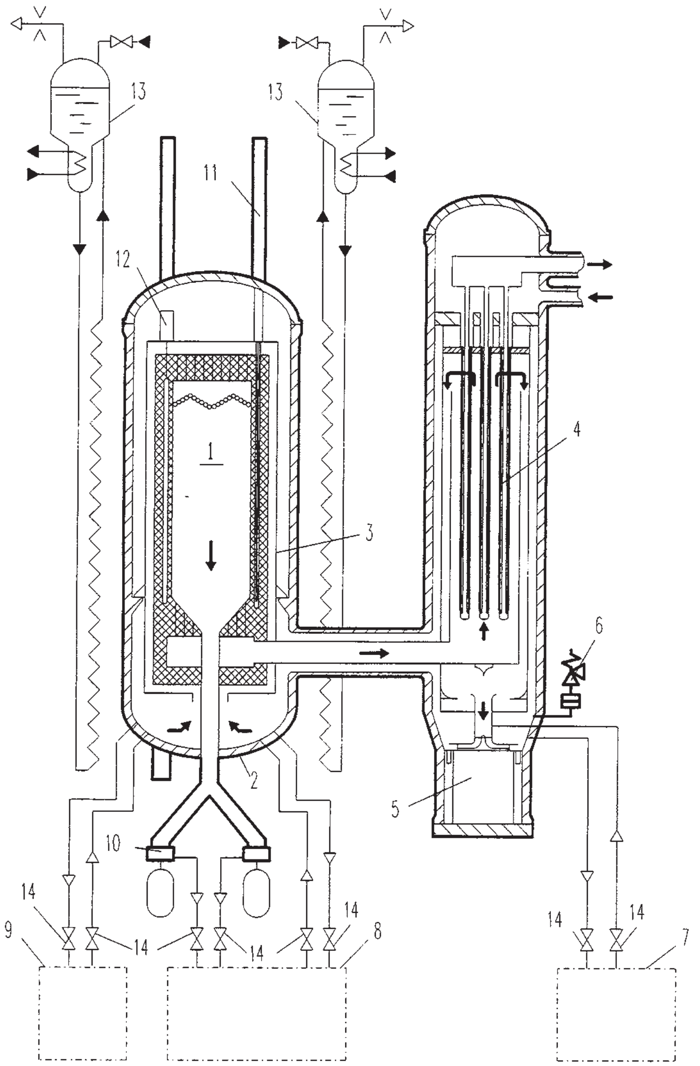
14.9.2 HTR-10

The Chinese are testing PBMR technology within the HTR-10 project. HTR-10 is a small test reactor of only 10 MWt and is operated by the Chinese Institute of Nuclear Energy Technology (INET). The reactor first went critical in late 2000. The fuel has been fabricated in China but is based on German fuel technology. The thermal–hydraulic cycle is being tested in several stages. Initially the steam/power cycle is being verified. This will be followed by testing of the gas turbine cycle.

The Institute of Nuclear Technology (INET) in Beijing, China, is developing several reactor systems for non-electrical applications (Sun *et al*., 1998). Technologies for water-cooled heating reactors and for modular high-temperature reactors are being developed. A 5-MW water-cooled test reactor was constructed in 1989 and feasibility studies for seawater desalination using the reactor as the power source are in progress. For high-temperature applications, a 10 MWt test reactor, HTR-10, is seen as a step towards developing a commercial HTGR demonstration plant.

14.9.3 BN-600 Application for SLF Production

IPPE, Obninsk and the Combustible Resources Institute (CRI) have proposed a project using fast neutron reactors such as BN-600 for power to be used for *SLF production from low-grade coals or heavy petroleum. The intention of the scheme is to use the existing technology of the BN-600 plant. The process will involve fuel hydrogenation using



1 - core; 2 - vessel system; 3 - leak tight core shell; 4 - intermediate heat exchanger;
5 - primary circuit circulator with cut-off valve; 6 - pressurization protection device;
7 - primary coolant purification system; 8 - refueling complex; 9 - small absorber balls system;
10 - fuel element unloading mechanism; 11 - control rod drive; 12 - small absorber balls system drive;
13 - emergency cooldown system; 14 - localizing valves.

Figure 14.4. VGM-P reactor. Source: Golovko *et al*. (1998).

technologies developed by CRI and take advantage of previous worldwide experience from the UK and Germany (Troyanov *et al.*, 1998). The technological processes for SLF production are described in Mourogov *et al.* (1994).

BN-800 is also proposed as a 800 MWe version.

14.10. HYDROGEN PRODUCTION

Of all the high-temperature applications, there is probably maximum interest in hydrogen at the present time. Hydrogen production is an important long-term objective of the Generation IV Programme.

The challenge is to develop a hydrogen generation process that does not release greenhouse gas such as carbon dioxide (Institute of Nuclear Engineers, 2004). The classical fossil fired steam reformation of methane has this problem and methods of reducing the CO_2 release are under development. Other techniques being investigated include high-temperature electrolysis and also thermo-chemical water splitting. Neither of these methods produce CO_2.

The generation of hydrogen using nuclear heating is under consideration within the JAERI HTTR programme described below. Hydrogen production technology is also being considered within the US Next Generation Nuclear Plant programme at Idaho.

14.10.1 HTTR

The high-temperature test reactor (HTTR) finished construction in 1996 (Figure 14.5). The power rating was 30 MWt and first criticality was achieved in 1998. This reactor includes the annular prismatic fuel design. The core outlet temperature is currently 850°C but may be increased by 100°C following design optimisation. This reactor is one of kind envisaged for process heat applications and, therefore, includes an intermediate heat exchanger with the purpose of supplying process heat.

Research is being conducted at JAERI on the high-temperature engineering test reactor (HTTR) for heat utilisation (Miyamoto *et al.*, 1998). This is the first high-temperature gas reactor (HTGR) to be constructed in Japan. The design is for a 30-MW thermal output and outlet coolant temperature of 950°C. After a satisfactory demonstration period, a hydrogen production system will be fitted. The process will involve steam reforming of natural gas (Hada *et al.*, 1996). It has been demonstrated in out-of-pile tests at 1/30 scale carried out by the Science and Technology Agency (Inagaki *et al.*, 1997).

Nuclear heat of 10 MW at 950°C is supplied from the HTTR to a heat exchanger in a primary helium loop. A secondary helium loop then transfers heat to the steam reformer, which converts steam and methane to hydrogen and carbon monoxide. To provide stability

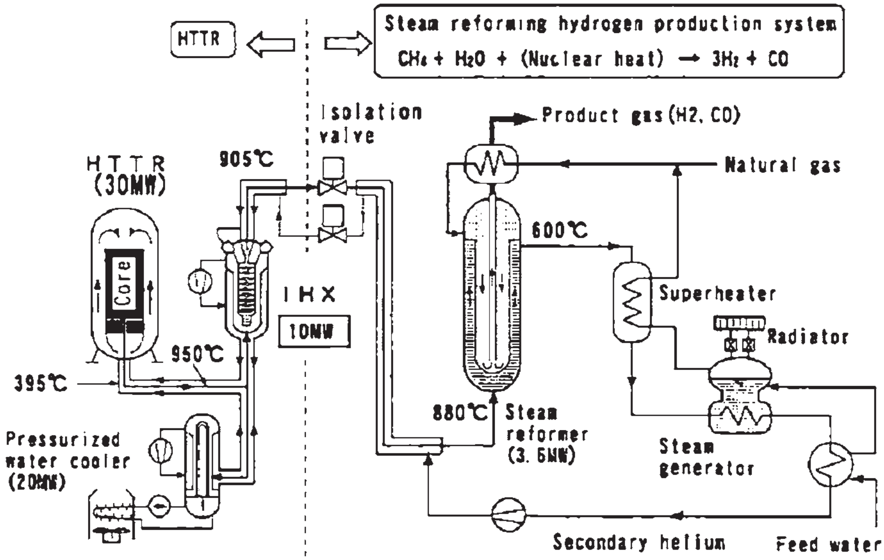


Figure 14.5. HTTR hydrogen production system. Source: Miyamoto *et al.* (1998).

in the event of disturbances in the steam reforming process, a steam generator is installed at the downstream of the steam reformer to keep the helium gas temperature at the steam saturation temperature.

To reduce carbon emissions, further studies are in progress on hydrogen production by water splitting, via a thermochemical iodine sulphur process first proposed by the General Atomic Company (Norman *et al.*, 1982). This is foreseen as an improved potential heat utilisation and hydrogen production process for the HTGR.

In addition to hydrogen production, there are other high-temperature applications being proposed, including the production of gases such as styrene and ethylene.

14.10.2 Generation IV Systems

These have been considered in Chapter 12.

14.11. OTHER TOPICS

Finally, there are some possible very high-temperature applications that could be conceivable with the VHTR designs that are the end objective of Generation IV. These include iron manufacture, cement and even glass process applications (Institute of Nuclear Engineers, 2004).

REFERENCES

Achkasov, A., Grechko, G. & Shishkin, V. (1997) NIKA-120M nuclear steam supply system with integral type reactor for ground and floating NPPs, *Nucl. Engng Des.*

Adamov, E.O. & Romenkov, A.A. (1996) The project of apatity nuclear heating plant. Modern technical and economic issues of nuclear heat application in Russia, the experience with nuclear heat applications: district heating, process heat and desalination, IAEA-AGM-915, IAEA-CT 2967.

Adamov, E.O., Cherkashov, Yu.M. & Romenkov, A.A. Non-electric application of pool-type nuclear reactors, non-electric applications of nuclear energy, *Proceedings of an Advisory Group Meeting held in Jakarta, Indonesia, 21–23 November 1995*, IAEA-TECDOC-923, 1997.

Adamovich, L., Grecho, G. & Shishkin, V. (1997) Self-contained co-generation nuclear power plant with integral nuclear reactor designed for remote and difficult-to-access areas, *Nucl. Engng Des.*

Awerbuch, L. (1997) Power desalination and the importance of hybrid ideas, *Proceedings of the IDA World Congress on Desalination and Water Re-use, Madrid*.

Baranaev, Yu.D., Osennikh, G.M., Sergeev, Yu.A., Mikhan, V.I. & Romenkov, A.A. (1998) Project of demonstration nuclear heating plant using pool-type water reactor, IAEA-TECDOC-1056, November 1998.

Chang, M.H. & Kim, S.H. (1998) Approach for SMART application to desalination and power generation, IAEA-TECDOC-1056, November 1998.

Chernozoobov, V.B., Tockmantsev, N.K. & Putilin, Y.V. (1995) From the experience of development and exploration of large distillation desalination facilities, IAEA Technical Committee Meeting, Obninsk, Russia, May 1995.

Csik, B.J. & Kupitz, J. (1997) *Nuclear Power Applications: Supplying Heat for Homes and Industries*, IAEA Bulletin, **39 (2)**, IAEA, Vienna.

Dazhong, W., *et al*. (1996) Research and development of nuclear heating reactors in China, *Tsinghua Sci. Technol.*, **1** (1), 1–7.

Duo, D. *et al*. (1995) Study of the 200MW NHR nuclear seawater desalination and interface plan, *Nucl. Power Engng*, **16** (4), August 1995.

Golovko, V.F., Kiryushin, A.I., Kodochigov, N.G. & Kuzavkov, N.G. (1995) State of HTGR development in Russia, *Proceedings of a Technical Committee Meeting held in Beijing,* China, 30th October–2nd November 1995.

Golovko, V.F., Guchshin, U.L., Kodochigov, N.G., & Kuzavkov, N.G. (1998) Utilisation of HTGR heat and its transfer to industrial facilities, IAEA-TECDOC-1056, November 1998.

Grechko, A.G., Romenkov, A.A. & Shishkin, V.A. (1998) Technical and economic evaluation of nuclear seawater desalination systems, IAEA-TECDOC-1056, November 1998.

Gromov, B.F., Orlov, Yu.I., Toshinsky, G.I. & Chekunov, V.V. (1996) Nuclear power facility with lead–bismuth coolant, ISSN 0004-7163. *Nucl. Energy*, **81**, 340–347.

Hada, K., Nishihara, T., Shibata, T. & Shiozawa, S. (1996) Design of a steam reforming system to be connected to the HTTR, *Proceedings of the Third JAERI Symposium on HTGR Technologies*, JAERI Conf. 96-010.

Hanra, M.S. & Misra, B.M. (1998) Desalination demonstration plant using nuclear heat, IAEA-TECDOC-1056, November 1998.

Humphries, I.R. & Davies, K. (1995) A floating generation system using the Russian KLT-40 reactor and Canadian reverse-osmosis water purification technology, IAEA Technical Committee Meeting, Obninsk, Russia, May 1995.

IAEA (1998) Nuclear desalination of sea water, *Proceedings of the Symposium Taejon, 1997*, Vienna.

IAEA-TECDOC-881 (1996) Design and development status of small and medium reactor systems, 1995, Vienna.

IAEA-TECDOC-923 (1997) Non-electric applications of nuclear energy, IAEA, Vienna.

IAEA-TECDOC-1056 (1998) Nuclear heat applications: design aspects and operating experience, November 1998.

IEA/OECD (NEA)/IAEA (2002) Innovative nuclear reactor development: opportunities for international co-operation, 'Three Agency Study'.

Inagaki, Y., Hino, R., Hada, K., Nishihara, T., Takeda, T. & Shiozawa, S. (1997) Out-of-pile demonstration programme of HTTR hydrogen production by steam reforming of natural gas, *Proceedings of the Fifth Conference on Nuclear Energy*, ICONE 5-2342.

Institute of Nuclear Engineers (2004) Generation IV seminar on nuclear energy systems research and development, Institute of Nuclear Engineers, DTI, London, March 2004.

Kim, S.H. & Chang, M.H. (1997) Korea's R&D programme and activities for nuclear desalination, IAEA-SM-347/9, Symposium on Desalination of Seawater with Nuclear Energy, Taejon, Korea, May 26–30, 1997.

Komkova, O.I., Kononenko, G.J., Puchkov, P.V. Sidorov, V.I., Trevgoda, M.M. & Verhovodko, S.Z. (1998) Development of small (10 MW thermal) nuclear plant with lead–bismuth coolant for electricity and heat co-generation, production of fresh water and hydrogen, IAEA-TECDOC-1056, November 1998.

Miyamoto, Y., Shiozawa, S., Ogawa, M., Akino, N., Shimizu, S., Hada, K., Inagaki, Y., Onuki, K., Takeda, T. & Nishihara, T. (1998) Overview of HTGR heat utilisation system development at JAERI, IAEA-TECDOC-1056, November 1998.

Mourogov, V.M., Troyanov, M.F., Sidorov, G.I. *et al.* (1994) Creation of the project of typical power technology complex of ecologically compatible production of synthetic liquid fuel (SLF) from coals and heavy petroleum residues, IPPE, Obninsk, RF. Obninsk Symposium of XV-th Mendeleev Congress on General and Applied Chemistry, Part 1.

Norman, J.H., Besenbruch, G.E., Brown, L.C. & O'Keefe, D.R. (1982) Thermochemical water-splitting cycle, Bench Scale Investigations and Process Engineering, GA-A16713, 1982.

Panov, Y.K., Polunichev, V.I. & Zverev, K.V. (1998) Nuclear floating power desalination complexes, IAEA-TECDOC-1056, November 1998.

RDIPE (1994) Conception of perspective autonomous energy source for heat supply, RUTA-TE, Moscow.

Seo, J.K. *et al.* (1997) Advanced integral reactor (SMART) for nuclear desalination, IAEA-SM-347/40, Symposium on Desalination of Seawater with Nuclear Energy, Taejon, Korea, May 26–30, 1997.

SSC RF-IPPE, EDB (1996) A technical possibility and economic feasibility of the NVNPP 2nd, 3rd, and 4th units renewal when their service life is exhausted, SSC RF-IPPE, EDB Gidropress and Atomenergoproekt, TECDOC, Obninsk, 1996.

Stepanov, V.S., Kulikov, M.L., Ilyin, O.G., Klimov, N.N., Chekunov, V.V., Grigoriev, O.G., Gromov, B.F., Leonchyk, M.P., Pashkin, Yu.G., Pankratov, D.V., Toshinsky, G.I. & Skorikov, D.E. (1998a) SVBR-75: a reactor module for renewal of VVER-440 decommissioning reactors – safety and economic aspects, IAEA-TECDOC-1056, November 1998.

Stepanov, V.S., Legyuenko, S.K., Grigoriev, O.G., Gromov, B.F., Dedul, A.V., Leonchyk, M.P., Pashkin, Yu.G., Pankratov, D.V., Chekunov, V.V., & Skorikov, D.E. (1998b) The ANGSTREM project: present status and development activities, IAEA-TECDOC-1056, November 1998.

Sun, Y., Zhong, D., Dong, D. & Xu, Y. (1998) Nuclear reactor development in China for non-electrical applications, IAEA-TECDOC-1056, November 1998.

Troyanov, M.F., Popavskij, V.M., Sidov, G.I., Bondarenko, A.V., Chebeskov, A.N., Chushkin, V.N., Karabash, A.A., Krichko, A.A., & Maloletnev, A.S. (1998) Power technology complex for production of motor fuel from brown coals with power supply from NPPs, IAEA-TECDOC-1056, November 1998.

Wangnick, K. (1995) The historical development of the desalination market, IDA World Congress on Desalination and Water Sciences, Abu Dhabi.

Zhang, Y., Zhang, D. & Dong, D. (1998) Analysis of the nuclear heating reactor and its possible application in seawater desalination, IAEA-TECDOC-1056, November 1998.

Zheng, W. (1998) Design precautions for coupling interfaces between nuclear heating reactor and heating grid or desalination plant, IAEA-TECDOC-1056, November 1998.

Zuznetsov, Yu.N., Lisitsa, F.D., Tokarev, Yu.I. & Romenkov, A.A. (1998) Nuclear power plant with pressure vessel boiling water reactor VK-300 for district heating and electricity supply, IAEA-TECDOC-1056, November 1998.

Chapter 15
Experimental Research Programmes

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Chapter 15

Experimental Research Programmes

15.1. INTRODUCTION/OBJECTIVES

In connection with design optimisation, comprehensive experimental and theoretical programmes for component and integral design, testing and certification are carried out by vendors. In addition, safety-related projects are performed by major research institutes within various national and international programmes. Many data available for present generation plants are still applicable to evolutionary and advanced future reactors. However, additional data are required for certain physical processes which occupy increased significance in advanced passive systems, e.g. natural circulation, condensation and non-condensable gas-related phenomena. Relevant experimental research for both current and future reactor systems is summarised for some of the major designs under review in this book. In regard to international activities, particular reference is made to the extensive EC research programmes (FISA 2001, 2001; FISA 2003, to be published). The focus in this chapter will be on experimental programmes; theoretical work is covered in the next chapter.

The majority of research to date has been in support of the present generation and evolutionary plant. Many present day plants are coming up towards the end of their original design lives and with the dearth in new build there are strong incentives to consider extension of life. This has, therefore emerged as a key area of research. Severe accident research is being used to develop severe accident plans and support Level 2 PSAs; hence severe accident research is also prominent. Research specific to evolutionary passive designs includes work on natural convection cooling, with and without the presence of water. Many of the evolutionary designs also include improved provision against severe accident loads; research is carried out on passive heat removal from melts within the reactor cavity.

Significant areas of research to support the present generation programme include safety at all stages of the fuel cycle, reactor safety during plant operation, radioactive waste management, radiological protection, and other activities to benefit from ‘lessons learned’ in the past. There are active work programmes in all these areas; for example within the European Union there have been numerous activities funded by the EC Euratom Programme and corresponding counterpart national programmes.

In addition to research projects *per se*, there are a number of joint and other collaborative projects being co-ordinated under the auspices of the NEA, which primarily collect, make data available and perform analysis on the data. Some of these projects are of a general nature, e.g. the International Common-cause Data Exchange (ICDE) project collects operating data related to common-cause failures. The Fire Project collects data related to fire events in nuclear environments and the OECD Piping Failure Data Exchange (OPDE) project collects and analyses pipe failure event data. More information is given in NEA Annual Report (2002).

There are a number of technical issues that will need to be addressed in regard to the innovative reactor systems that are envisaged for the future, e.g. the Generation IV concepts. These are covered briefly at the end of this chapter. Many of the concepts will require significant research effort. For some systems, R&D activities are already underway, e.g. in regard to the supercritical and high-temperature gas systems that are expected to be among the first Generation IV systems to become available.

In the first part of this chapter, research primarily relevant to current generation plant is considered.

CURRENT GENERATION REACTORS

15.2. PLANT LIFE EXTENSION

The technical issues associated with plant ageing centre around the ageing of mechanical, electrical and materials ageing of plant components, particularly concretes and steels (Govaerts, 2001). The EC is funding a major research programme on this issue and a selection of some of the on-going projects is summarised below (Table 15.1). Utility practices for the safe management of nuclear power plant ageing in the EU are given in

Table 15.1. EC Research in nuclear fission energy (1998–2002)

Operational safety of existing installations	Plant life extension and management Severe accident management Evolutionary concepts
Safety of the fuel cycle	Waste and spent fuel management and disposal Partitioning and transmutation Decommissioning of nuclear installations
Safety and efficiency of future systems	Innovative and revisited concepts
Radiation protection	Risk assessment and management Monitoring and assessment of occupational exposure Off-site emergency management Restoration and long-term management of contaminated environments

FISA 2001: EU Research in Reactor Safety (2001) and FISA 2003 (to be published).

Table 15.2. Plant life extension and related issues

Issues	EC research programmes
Embrittlement of materials	AMES, PISA, FRAME, RETROSPEC, GRETE
Materials corrosion	PRIS, INTERWELD
Fracture mechanics	NESC, SMILE
Concrete ageing	MAECENA, CONMOD
Materials testing	FEUNMARR
Thermal–hydraulics	WAHALOADS

FISA 2001: EU Research in Reactor Safety (2001) and FISA 2003 (to be published).

EUR 19843 (2001). The phenomena include thermal fatigue and stress corrosion, and relate to the thermal and mechanical loads to which the components are subjected. Chemical factors may also be an issue.

There may also be practical factors that present a range of difficulties in presenting a case for life extension; e.g. hardware and software may become obsolete, original suppliers may no longer be able to supply replacements, etc. Computer codes may become outdated and no longer supported by developers. Rules and standards may change. Knowledge may reside in staff who have retired or about to retire, documentation may not be adequate without the presence of experienced original authors. Modern non-destructive testing (NDT) methods may be able to identify defects that had not previously been observed, but also in a positive sense, may be able to confirm the absence of defects (Table 15.2).

15.2.1 Embrittlement of Materials

The ageing materials European strategies (AMES) network was set up by the EC to bring together expertise on nuclear reactor materials (Gerard *et al*., 2001; Sevini *et al*., 1999). The most important area for research for effective plant life extension and management is the reactor pressure vessel (RPV), but metallic components in general (e.g. internals, pressuriser and piping) were targeted in AMES. The other principal areas are irradiation embrittlement and thermal ageing. In recent years, the network has been enlarged to include representatives from the Central and East European countries. AMES members collaborate in the TACIS and PHARE programmes to integrate findings for PWR and VVER LWRs.

The phosphorus influence on steel ageing (PISA) programme (English *et al*., 2001) is an experimental study to investigate the influence of phosphorus on RPV steel irradiation embrittlement. The objective is to improve understanding by segregating the phosphorus to grain boundaries and determining the effect of brittle inter-granular failure mechanisms on the RPV properties. The experiments focus attention on investigating various irradiated steels and metal alloys. The lack of phosphorus segregation data on certain steels under irradiation conditions relevant to end-of-life was recognised in a recent review

(English *et al*., 2002). Both PWR and VVER reactor designs are covered in the project. The understanding of phosphorus segregation in irradiated and thermally aged fuels is now advancing significantly.

Another EC programme, fracture mechanics based embrittlement (FRAME) (Valo *et al*., 2001) aims to irradiate a relatively large number of different materials, chosen to determine the effects of chemistry on embrittlement. The objective is to develop fracture mechanics based trend curves. Irradiation shifts are measured and these are compared with existing Charpy-V (CH-V) based regulatory and other trend curves. Since the cleavage initiation fracture toughness material property K_{JC}, is required for pressurised thermal shock (PTS) safety analyses, the availability of directly measured data will help to remove uncertainties (Sokolov and Nanstad, 2000) from the utilisation of CH-V test data.

The need for accurate data on neutron fluence to be used in conjunction with materials data is important for determining the life of nuclear power plant components, particularly the RPV. The RETROSPEC Dosimetry programme of the EC (Voorbraak *et al*., 2001) aims to provide retrospective fluence data by focussing on the niobium reaction $^{93}Nb(n,n')^{93}Nb^{m}$. The methodology is being developed by examining specimens from material test programmes in research reactors, the Petten High Flux Reactor (HFR) and from specimens in surveillance capsules from the Dukovany NPP and the Loviisa NPP. Four steels have been selected, which are representative of the RPV in East European VVERs. The methodology is validated by comparing results from the retrospective analysis with the measured fluence at the locations of the specimens. It is concluded that retrospective dosimetry is useful in determining the neutron fluence at various locations inside a nuclear reactor, e.g. at RPV welds. Retrospective dosimetry has been reported previously by a number of researchers, see also van Aerle *et al*. (2000).

The EC GRETE programme (Delnondedieu *et al*., 2001) is concerned with the development of innovative non-destructive techniques for the inspection of critical components that may affect decisions on the lifetime of the plant. The objective is to assess techniques that aim to detect changes in materials before macro-structural defects occur, thus allowing remedial action to be taken. The techniques are evaluated in relation to neutron irradiation damage of the reactor vessel and the thermal fatigue of piping of the primary loops. Aged samples are being examined metallurgically and mechanically and then tested using various non-destructive techniques. All the known NDT techniques and their limits and limitations have been listed within the frame of the AMES project, see Delnondedieu *et al*. (2001) and Series of AMES reports (1975).

15.2.2 Materials Corrosion

High fast neutron fluence in RPV internals can change the ductility and fracture resistance of the material. Cracking has been detected in some RPV internal components such as the core shrouds and top guides of BWRs and this has resulted in the need for more data on the irradiated material properties. A concern has also been expressed as to whether high neutron doses could cause void swelling and, therefore embrittlement induced by the voids. These phenomena could clearly impact on the life of a plant. In order to address these issues, the EC PRIS project has been set up to examine the properties of irradiated stainless steels for predicting the lifetime of such nuclear power plant components (Nordgren *et al.*, 2001).

The project involves the procurement of representative top parts of BWR control rods blades of type AISI 304L and type AISI 316L stainless steel with fast neutron fluences in the range 2×10^{21}–5×10^{21} n cm^{-2}. These specimens are being examined, mechanical properties are being determined and the microstructure is being characterised.

A thimble tube of type AISI 316 stainless steel from the Swedish Ringhals 2 plant that has been irradiated for 23 years to between 0 and 70 dpa is also being examined. Tensile and hardness properties, fracture properties and radiation-induced micro-structural and micro-chemical changes will be determined. Fracture properties will be determined using previously established pin-loading fracture toughness test techniques (Grigoroev *et al.*, 1995, 1997).

The properties of both the BWR- and PWR-irradiated materials will be compared with non-irradiated archive materials.

Stress corrosion cracking in PWR and BWR shroud internals are also under study in the EC INTERWELD project (Youtsos *et al.*, 2001). The objective of this project is to define better the radiation-induced material changes in the heat-affected regions of austenitic stainless steels.

Test welds of stainless steel type 304 and type 347 are being produced with weld residual stresses, microstructure and properties that are representative of core shroud applications. These are being irradiated to two neutron fluence levels in the HFR at Petten, the low level at 0.3 dpa and the high level in the range 0.8–1.2 dpa. These levels are representative of LWR internal irradiations. The results will be compared with an in-service weld from the BR3 reactor. This weld has been irradiated from 1962 to 1987 in a coolant of temperature 260–300°C with maximum dose irradiation of 2.4×10^{20} n cm^{-2}.

The weld residual stresses of the irradiated materials are being measured by neutron diffraction and the corrosion characteristics of the material will be determined by further tests. Mechanical properties are being determined for both the test specimens and the in-service material. The microstructure and microchemistry properties are being obtained by optical, EPMA and other techniques.

15.2.3 Fracture Mechanics

In 1993, a network for the evaluation of structural components (NESC) (Rintamaa and Taylor, 2001) was formed, based on a multi-partner collaboration agreement and managed by the Joint Research Centre at Petten. It was composed of utilities, regulators and research organisations and the main objective was to develop and validate structural integrity techniques for assessment. There was a broad representation of countries across Europe, covering countries operating a wide range of nuclear power plants.

There have been four NESC projects.

NESC-1 was the first large-scale project to evaluate the whole process of structural integrity assessment. In particular, the spinning cylinder PTS test was designed to simulate the conditions of an ageing RPV subjected to a severe PTS loading. It demonstrated the beneficial effect of cladding in inhibiting cleavage initiation in the cylinder surface. It was used to validate structural mechanics assessment techniques and to validate no-destructive inspection techniques.

The NESC-2 programme included two large-scale PTS tests on thick wall (200 cm) cylinders with shallow defects. The objective of the tests was to consider brittle crack initiation, the propagation and arrest of shallow cracks in a cladded vessel under PTS loading. The first test, which included a circumferential under-clad notch of depth 8 mm, exhibited a crack growth that was arrested. In the second test, there were two shallow semi-elliptical through clad effects but no growth occurred.

In NESC-3 there is a large-scale test on a dissimilar weld pipe assembly of aged PWR Class 1 piping. It is a benchmark test to demonstrate the load to cause failure at a large defect. The purpose is to quantify the accuracy of assessment procedures for a defect containing dissimilar metal welds, to address issues regarding inspection performance, and to promote best practice.

The NESC-4 test series is to test defect-containing beams, designed to clarify the role of bi-axial stress effects on shallow flaws in RPV weld material.

There are several other EC R&D programmes that have links with NESC. These include exploratory or pilot projects that investigate certain aspects that could lead to further major tests (Tice *et al.*, 1999; Faidy *et al.*, 2000; Leggart *et al.*, 1999) and other collaborations that utilise NESC results (Lidbury *et al.*).

The European SMILE project (Bezdikian *et al.*, to be published) considers whether the structural margins of aged embrittled RPVs can be improved if a particular potential beneficial effect of load history is taken into account (warm pre-stress). The programme will provide data from representative steels.

In recent observations, different cracks have been discovered in different US and European nuclear power plants (VC SUMMER, RINGHALS, BIBLIS). The issue is the integrity of aged cracked metal welds involving different materials, e.g. ferritic to stainless steel. The extent of crack growth and paths followed by a crack through the weld will be followed under various loads in the ADIMEW project (Faidy *et al.*, to be published).

15.2.4 Concrete Ageing

There are now many nuclear power plants operating which are at least 30 years old and many are approaching the end of their original design life. Central to the continued safe operation of these plants is the structural integrity of various safety critical components.

One such is the concrete pressure vessel. These vessels have to withstand large internal pressures ($\sim$4 MPa at 700°C in the case of an UK AGR). During lifetime, the pressure vessels deform and age.

The MAECENA project (Crouch *et al.*, to be published) has the objective of investigating an important area of concrete behaviour that influences the ageing process, i.e. the softening and weakening of the effects of thermal and pressure cycling, and progressive creep and relaxation. The programme involves laboratory-based experimental work together with the development of finite element code methodology.

Concrete containment behaviour under various loading conditions has been considered in the European Commission CONMOD programme (Jovall *et al.*, to be published). This aims to create a system for the assessment of containments throughout their lifetime. An important aspect is to develop NDT techniques and integrate these with finite element modelling techniques.

15.2.5 Materials Testing

European materials test reactors (MTRs) have successfully served the nuclear power industry over many years. They have fulfilled many different roles in supporting nuclear power development; additionally they are well known for their role in the production of radio-isotopes for medical applications. Successful reactors include: BR2 in Belgium, R2 in Sweden, HFR in the Netherlands and LVR15 in the Czech Republic. However, by 2010 all these reactors will have reached 50 years and, therefore reaching the end of their operational life. There is an important European initiative FEUNMARR (Parrat *et al.*, to be published) to address this issue and define the needs for future MTRs.

15.2.6 Thermal–Hydraulics

The loads on equipment and structures in nuclear power plants due to water hammer phenomena are being examined as part of the EC 5th Framework Programme WAHALOADS (Giot *et al.*, 2001). The main interest is in water hammer due to condensation or shock waves. This might be caused by the inflow of sub-cooled water into pipes or other components containing steam or two-phase steam–water mixtures. Pressure waves might be generated by valve operation or following pipe ruptures.

Water hammer data are being obtained from three different test facilities.

The UMSICHT facility in Oberhausen is being adapted to simulate pipes with supports, in a configuration that is prototypic of a nuclear power plant. Experiments are being conducted with the opening and closing of valves in two 230 m pipes at different elevations, the pipes have inner and outer diameters of 54 and 108 mm, respectively. Fluid dynamic loads, fluid structure interactions and global structural response will be investigated.

The Cold Water Hammer Test Facility in FZ-Rossendorf aims to generate water hammer by accelerating a water slug to impinge on a lid flange (bouncing plate). The facility consists of a pressurised water tank connected to a horizontal pipe section connecting through a 90-degree bend to a vertical pipe section with the lid flange. The total length of the pipe is 3 m with outer diameter 219 mm.

The water hammer test rig in the integral test facility PMK-2 at AEKI will be used to perform at system pressures up to 4 MPa. A horizontal pipe of 80 mm inner diameter is connected to the head of the steam generator on one side and the steam condenser of the facility on the other side. Water hammer is generated by displacing steam in the test pipe with cold water.

CURRENT GENERATION AND EVOLUTIONARY REACTORS

15.3. FUEL BEHAVIOUR

This section examines the experimental programmes devoted to fuel behaviour under normal operation but focussing particularly on transient/accident conditions. Examples of important current programmes are given in Table 15.3. Research is progressing on the behaviour of advanced fuels and clads in current generation plant to optimise performance, without challenging the margins of existing safety cases. Much of the present work is also relevant to evolutionary water designs. Many of the issues were discussed earlier in Chapter 5. Research programmes for advanced fuel cycles play an important part in the progress towards the future innovative designs that are being considered under the Generation IV initiative. These are discussed later in this chapter.

15.3.1 Normal Operation

Many of these data are proprietary to fuel vendors and the data are not publicly available. Relative fuel bundle performance in regard to grids and safety margin is an important differentiator across different designs. Nevertheless, some data are available from international bodies, e.g. the NEA data bank maintains an international fuel performance experiments (IFPE) database which contains a wide range of fuel performance data (NEA Annual Report, 2002).

15.3.2 Transient/Accident Conditions

15.3.2.1 Argonne National Laboratory (ANL). The USNRC, with the co-operation of EPRI, are sponsoring an experimental test programme at ANL (Fuel Safety Criteria

Table 15.3. Fuel behaviour

Issues	Experimental programmes
Normal operation	Vendor proprietary programmes, IFPE
Transient conditions	ANL, CABRI, Halden, NSRR

Fuel Safety Criteria Technical Review (2000), Bassette (2000), Papin and Schmitz (1999), Wiesenack (1997), Fujishiro and Ishijima (1994), Fuketa (1999).

Technical Review, 2000; Bassette, 2000) to determine the behaviour of high burn-up fuel under simulated LOCA conditions. Another primary objective is to provide data on the mechanical properties of high burn-up cladding for analysing the transients of interest in safety case analysis.

Secondary objectives are to develop a methodology for estimating fuel behaviour under LOCA conditions that can be applied to different cladding types of similar properties. There are also benchmark tests on fresh cladding to determine low-burn-up properties on modern day clads and to check for consistency with earlier results.

Three types of tests are being conducted – oxidation tests, to develop and validate kinetics models; quenching tests to evaluate current acceptance criteria or for establishing new criteria; and also structural response tests to establish whether coolable geometry or control rod insertion could be affected by external mechanical loads.

15.3.2.2 CABRI. The CABRI reactor is managed by the Institute for Radiological and Nuclear Safety (IRSN), previously the Institute for Protection and Nuclear Safety (IPSN) in France. The main purpose of the facility is to test the response of fuel rods under reactivity-initiated accident (RIA) conditions (Fuel Safety Criteria Technical Review, 2000; Papin and Schmitz, 1999). It consists of a water pool driver reactor and a sodium-cooled experimental loop. Fuel rods are subjected to pulses of a few tens of milliseconds, and the timescales of the rod temperature transient are very fast, representative of fast reactor RIA conditions. Tests have been conducted for both high burn-up UO_2 and MOX fuel. Where a fuel failure occurred in a MOX rod it is observed that events are more energetic than for the UO_2 case. It is thought that enhanced fission gas migration to the grain boundaries, combined with higher porosity in the Pu rich region of the MOX fuel, results in the greater fuel dispersion and coolant ejection in the MOX case.

The CABRI facility is being modified to include a water loop to create more representative PWR conditions. This will enable temperature transients on timescales several orders of magnitude to be examined, more representative of RIA timescales in a PWR. The thermal–hydraulic conditions are likely to have a much greater influence on events than in the sodium-cooled case. The CABRI water loop programme is being co-ordinated in the international community by OECD. A schematic is shown in Figure 15.1.

15.3.2.3 Halden. The Halden project in Norway has been running for many years and addresses many facets of fuel performance. It is co-ordinated by the OECD and is supported by various national and industrial bodies in about 20 countries. The Halden test facility incorporates high resolution and advanced instrumentation on rod thermal response, fission product gas release and pellet clad interaction (PCI) (Fuel Safety Criteria Technical Review, 2000; Wiesenack, 1997).

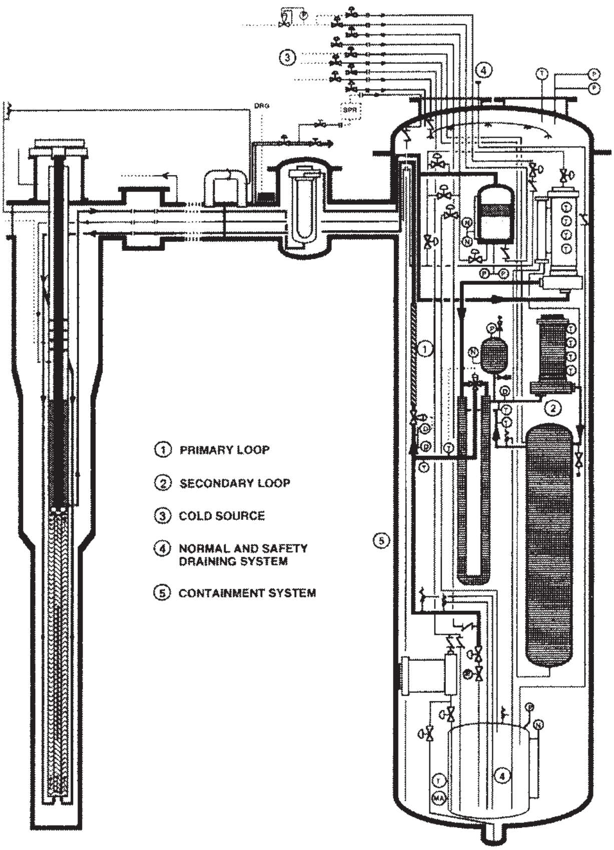

Figure 15.1. CABRI water loop. Source: IRSN (2002).

With the increasing emphasis on the performance of high burn-up fuel, rodlets of irradiated fuel from commercial reactors have been fabricated and tested for both normal and transient conditions. Burn-ups in ranges as high as 50–80 $\mathrm{MWd\,kg^{-1}}$ are included in recent test programmes. Tests can also be carried out with water loops, if appropriate.

The facility has also been used for testing the performance of MOX fuel and is a prime source of data for such advanced fuels.

15.3.2.4 NSRR. The Japanese Atomic Energy Research Institute (JAERI) in Tokai operates the Nuclear Safety Research Reactor (NSRR) for investigations of fuel rod performance under transient RIA conditions (Fuel Safety Criteria Technical Review, 2000; Fujishiro *et al*., 1994; Fuketa, 1999). The reactor is an annular water pool type reactor.

The recent programme has focussed on MOX fuel. For fresh fuel, the response of MOX fuel was found to be consistent with the behaviour of UO_2 fuel. Comparisons were made under similar test conditions. No effects of plutonium agglomerates were observed.

At increasingly higher burn-ups, the MOX fuel exhibited more fuel swelling and higher gas release than UO_2 fuel. These phenomena are consistent with the results from the CABRI programme.

15.4. REACTOR PHYSICS

Reactor physics related data are available in international data banks, e.g. the International Reactor Physics Benchmark Experiments (IRPhE) databank of the NEA and various activities are co-ordinated through the Nuclear Science Committee (NSC) (NEA Annual Report, 2002). The data are used as a reference for transient analysis to address specific safety issues. For example, a safety issue for PWRs concerns thermal mixing and the impact on neutronics. In 2002, a main steam-line break (SLB) benchmark study was carried out using data from the TMI-Unit 1 PWR. Coupled 3D neutronics/thermal–hydraulics calculational methods were used. For BWRs, reactor stability is an issue; benchmarks in 2002 have been performed based on US BWR-4 reference experimental data. Russian-designed VVER-1000 reactors have also been the subject of recent benchmark studies.

The NEA Data Bank services its member countries in regard to many requests for experimental and bibliographical nuclear data. Important improvements and updates are carried out through the joint evaluated fission and fusion (JEFF) activities. Data are available for ~ 340 different isotopes or elements including thermal scattering data for five lattice structures. The NEA is co-ordinating international collaboration among the major global nuclear data evaluation projects.

Experimental reactors are available for determining neutronic characteristics of reactor lattices, although the number of facilities is much less now in the world than hitherto. The EC is supporting the RENION project (Vasa, to be published) for carrying out PWR and VVER reactor physics related experiments.

15.5. THERMAL–HYDRAULICS

A large amount of thermal–hydraulic data has been identified and the data are available from the CSNI Code Validation Matrix for LWR LOCA and Transients (CCVM) database (NEA Annual Report, 2002). This is an important source of data because many of the large integral and separate effects thermal–hydraulics facilities are now closed; some remaining facilities that remain available are shown in Table 15.4.

Table 15.4. Thermal–hydraulics

Issues	Research programmes
Normal operation	Vendor programmes
Transient conditions	SETH (PKL, PANDA), FLOMIX-R, EREC, IMPAM

NEA Annual Report (2002), Weiss *et al*., to be published, Holmstrom *et al*., to be published.

Regarding recent experimental programmes to address current safety issues, the NEA/OECD SETH Project (NEA Annual Report, 2002) covers thermal–hydraulic experiments in support of accident management. Tests are being carried out at the Primar Kreislauf Loop (PKL) facility, owned by Framatome, to investigate boron dilution phenomena during a Small Break LOCA and also during mid-loop operation while the reactor is in a shutdown state.

The issue in the first case is whether low borated water slugs can form during a SBLOCA, which could then be driven into the core during restart of natural circulation, thereby resulting in a reactivity excursion. The second category is whether during a loss of residual heat removal (RHR) accident, conditions could exist for a boron dilution reactivity excursion.

The tests in the above programme were completed in 2002, including a series of LOCA tests and a mid-loop operation test. The tests indicated that boron dilution could be an issue that needs to be resolved by further tests.

A series of tests have been carried out in the PANDA facility in the Paul Scherrer Institute (PSI), Switzerland, within the SETH project. These experiments are to provide data on 3D gas flow in the containment on distribution and mixing issues. These data will be used for code validation, accident management and the development of mitigative measures. Condensation phenomena will also be addressed. PANDA is an ideal facility for application to evolutionary reactor research where natural circulation and strong coupling of the primary circuit and containment exists. The PANDA facility is referred to later in this chapter in the section on natural circulation in the context of evolutionary reactors.

Fluid mixing and flow distribution in the reactor circuit under conditions relevant to boron dilution, SLB and other transient conditions are under investigation in the FLOMIX-R project (Weiss *et al*., to be published). Experimental data on slug mixing have been obtained using high-resolution measurement techniques. Flow regimes cover highly turbulent flow and also buoyancy driven situations.

For VVER reactors there remain thermal–hydraulic and structural issues on the performance of bubbler condensers in VVER-440/213 reactors. These safety systems are designed to reduce the pressure in an LOCA situation. Experimental results are available from new experiments carried out at the Electrogorsk Research Centre (EREC) in Russia.

There has been a collaborative CSNI programme (NEA Annual Report, 2002) to analyse these data, comprising members from the EC, and regulators and utilities from the countries operating these reactors. Data for the Russian designed water reactor experiments have been incorporated into an extended CSNI Code Validation Matrix for LWR LOCA and Transients.

Thermal–hydraulic processes are also relevant to beyond design basis scenarios. These are being investigated in the context of VVER SBLOCAs in the EC project IMPAM-VVER (Holmstrom *et al.*, to be published) from an accident management point of view. The project also includes experiments from the Finnish PACTEL and Hungarian PMK-2 thermal–hydraulic facilities.

15.6. SEVERE ACCIDENTS AND THEIR MANAGEMENT

Severe accident research has been the main focus for reactor safety research over the past two decades. This largely started with the TMI-2 accident with phenomenological research, and the need to reduce severe accident risk further was re-enforced by the Chernobyl accident in 1986. For existing plants (Krugmann, 2001), measures have been introduced to reduce severe accident vulnerabilities, such as primary and secondary feed and bleed, filtered containment venting, hydrogen control by recombiners, igniters or by inerting, and filtration of control room air intake. For new designs, the IAEA has set more restrictive technical safety objectives (IAEA, 1999) such as severe core damage frequency less than 10^{-5} per plant operating year, elimination of sequences that could give rise to large early releases, and prevention of containment failure, thus limiting the need for off-site protection measures. These objectives have led to greater emphasis in reducing severe accident risk in the newer evolutionary designs.

There have been numerous research programmes over the past few decades to develop understanding of severe accident-related phenomena and also to develop guidelines for the prevention and mitigation of severe accidents. Much knowledge has been gained and at the present time, there is a reduction of effort on severe accidents R&D worldwide. Some research workers (Krugmann, 2001) believe that sufficient knowledge of severe accident phenomenology now exists. Confirmatory research, however, is still in progress in some areas. Also, clearly further work may be required to support a particular design in the event of new building. Recent research programmes are summarised in Table 15.5.

15.6.1 In-Vessel

The main areas for additional research on in-vessel related response, following a core melt event, relate to the timing and influence of reflood, both early and late (Krugmann, 2001). There are uncertainties in the late core degradation mechanisms and how these affect the pressure vessel failure mode. There have been experimental programmes at Sandia

Table 15.5. Severe accidents and their management

Issues	Research programmes
In-vessel core-melt	FOREVER, COLOSS, MASCA
Steam explosions	FARO, KROTOS, ECO, BERDA
Ex-vessel	ECOSTAR, FZK (DISCO), CEA, Argonne (MCCI project)
Source term	PHEBUS
Hydrogen and the containment	HYCOM, RUT Facility

Adroguer *et al*. (2001), Adroguer *et al*. (1999), Shepherd *et al*. (1999), Steinwarz *et al*. (2001), Jorge and Chaumont (2001), Seiler *et al*., Cognet *et al*. (1999), Steinwarz *et al*. (1999), WASH 1400 (1975), IRSN (2003), Benson *et al*. (1999) and Bechta *et al*. (2001).

National Laboratories, USA, the Paul Scherrer Institute (PSI), Switzerland and the Royal Institute of Technology (RIT), Sweden, addressing these issues. Particular EC projects in relation to vessel failure are the EC funded ARVI project and the FOREVER experiments at RIT. It has been shown that the pressure vessel failure mode impacts the integrity of the vessel supports, corium dispersal, missile generation and direct containment heating risk. In regard to outstanding issues, there is a research need to consider the hydrogen production rate in the event of delayed depressurisation as this impacts the hydrogen management control system. The composition and temperature of the gas discharge will depend on the response of the primary system discharge valves. At high pressure, the integrity of the SG tubes may also be an issue.

Activities currently in progress within the EC 5th Framework Programme include the following.

The core loss during a severe accident (COLOSS) (Adroguer *et al*., 2001) programme considers various issues concerning core degradation phenomenology. For both PWR and VVER rods, it includes the impact of UO_2 and ZrO_2 dissolution by molten Zircaloy on core geometry degradation. The objective is to examine the consequences on hydrogen production, melt generation and the source term. It also addresses how the burn-up effect affects the dissolution of UO_2 and MOX fuel by molten Zircaloy for PWR rods.

The experimental programme considers how the oxidation of U–O–Zr mixtures contributes to the peak hydrogen production during core reflood. Separate effects tests are carried out using a number of different composition U–O–Zr alloys. The results show that the oxidation of mixtures contributes to significant hydrogen release during degraded core quench.

Several large-scale tests are included to examine the B_4C effects, from absorber rods, on core degradation and melt progression. These include a large-scale VVER-1000 bundle test with a central B_4C rod, carried out in AEKI, Hungary, and a similar test with a B_4C rod carried out at FZK, Karlsruhe in Germany. Results show large escalation of oxidation and hydrogen during the final steam cooling phase, this phenomenon had not previously been observed.

The programme is also examining whether the oxidation of B_4C rods can induce volatile organic iodine production.

Some of these issues have been examined in earlier EC 4th FP projects, CIT (Adroguer *et al*., 1999) and COBE (Shepherd *et al*., 1999), which, respectively, were concerned with core material interactions and quench effects during core degradation.

Within the NEA collaborative programme, the MASCA (NEA Annual Report, 2002) project has also investigated the consequences of core melt within a severe accident. Experiments are being carried out in the Kurchatov Institute in which prototypical corium compositions are used. The experiments address the uncertainties on heat load to the reactor vessel and, therefore, on the uncertainties of vessel failure.

15.6.2 Steam Explosions

There has been considerable research over the years on whether steam explosions pose a risk to structural (containment) failure. Experimental programmes include FARO, KROTOS, ECO and BERDA (Jorge and Chaumont, 2001). There is evidence of pre-mixing of melt and water during the core relocation phase providing a mitigating effect. Experiments such as ECO (FZK) have shown that energy conversion factors now seem to be much lower than were originally envisaged. However, in, e.g., the French safety analysis process, in-vessel and ex-vessel steam explosion risks are still considered (Jorge and Chaumont, 2001). The R&D needs for in-vessel steam explosions are mainly concerned with gaining a better understanding of material effects and better characterisation of experiments (water, fuel and vapour fractions) (Seiler *et al*.). For ex-vessel melt, the main challenge is to establish the lack of explosive potential from realistic corium melt flow into water.

15.6.3 Ex-Vessel

Ex-vessel core melt phenomena have been studied to ascertain the feasibility of mitigation by water flooding or other means. The EC 5th Framework Programme ECOSTAR (Steinwarz *et al*., 2001) is concentrating on three important areas in relation to: melt release from the RPV, ex-vessel corium transport and long-term corium cooling. This programme builds on earlier projects CSC (Cognet *et al*., 1999), COMAS (Steinwarz *et al*., 1999) and CIT (Adroguer *et al*., 1999) to enhance the understanding of complex ex-vessel core melt behaviour, especially dispersion processes and jet formation, and their consequences.

To date, melt dispersion experiments using water/nitrogen fluids have indicated that lateral failures of the lower head lead to less melt dispersal out of the reactor cavity than do failures at the central part of the lower head. The new programme will examine the impact of fluid density on this conclusion. The erosion of different concretes with jets of iron melt and also oxide jets has also been studied. These experiments show that a metallic jet eroded the base-mat more deeply but that the oxide jet eroded a greater amount of the concrete. Melt dispersion experiments have been carried out in the DISCO facility at FZK, Karlsruhe and jet erosion is being studied in the KAJET facility, also at FZK.

Ex-vessel transport has been studied in the COMAS facility at the CARLA plant of Siempelkamp. This focuses on the spreading and distribution of the melt under molten core coolant interaction (MCCI) conditions. In a representative test, approximately 350 kg of oxidic melt are spread over a flat surface of siliceous concrete.

Reactor material experiments are in progress in regard to long-term stabilisation of the melt. Experiments with simulants have shown that phase segregation may exist within oxidic corium. Simulant experiments have been performed in the VULCANO facility at CEA using $ZrO_2(Al_2O_3$. Experiments have also been conducted in the ISABEL facility to determine the plane front solidification limits. Experiments are conducted to examine the efficiency of both top and bottom flooding as a means of cooling. Several series of experiments on dryout and quenching with different particulate beds have been conducted at KTK. A large-scale top flooding of a melt pool experiment has been carried out at Siempelkamp. Bottom reflooding is being examined at FZK in 1D and 3D.

The melt coolability and concrete interaction (MCCI) (NEA Annual Report, 2002) project at Argonne is managed by the USNRC and aims to provide experimental data on the spreading of molten debris over the base of the containment and the effectiveness of water cooling from the top. It also aims to provide information on the 2D interaction of the molten corium with the concrete structure of the containment, including the kinetics.

15.6.4 Source Term

Since the Rasmussen study in 1975, various potential containment failure modes giving rise to radioactive releases have been examined. Source terms have been identified at various stages according to the delay for containment failure and the potential for delayed release through some pathway with some possibility of retention, see for example WASH 1400 (1975).

In this case, the source terms have been classified into releases associated with:

- an early containment failure with a pathway for direct release;
- a delayed containment failure (24 h) with a pathway for direct release;
- a delayed release through a pathway including some radionuclide retention.

Experimental studies are being undertaken within the EC 5th Framework programme to quantify fission product and core materials released from molten corium during the late phase of a severe accident. This would be at a time when the integrity of the containment vessel might be threatened. This work has been carried out within the PHEBUS programme. A schematic of the facility is shown in Figure 15.2.

In addition to promoting understanding, another important objective of the PHEBUS programme is to provide well-instrumented data for the validation of integral severe accident computer codes. The main processes that effect the degradation of fuel

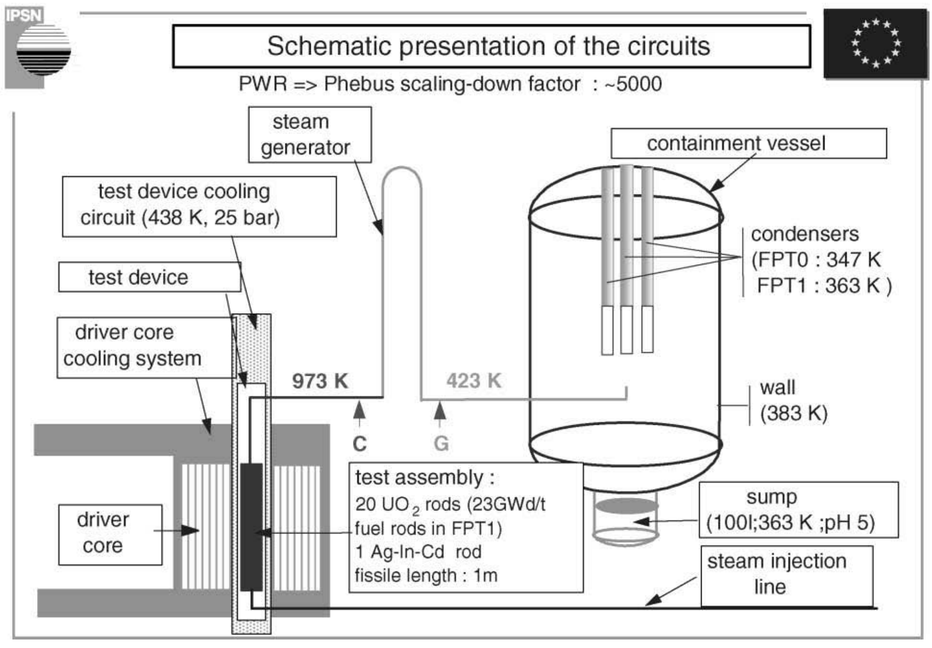

Figure 15.2. PHEBUS FP. Source: IRSN (2003).

and control rods, the release of fission products and aerosols, their transport in the primary circuit and the source term to the containment, are all included within the scope of the experiments.

Generally a good understanding has been gained of the releases of the more volatile fission products from intact fuel. However, the database for the release of less volatile fission products, core materials, aerosols, etc. from a degraded core is much less complete. This is particularly true for releases from a molten core. Previous experiments (Benson *et al*., 1999) have partially improved the database for the behaviour of metallic and ceramic pools. Additional data are required on the effects of fission product release in sparging and on the formation of crusts.

A current EC experimental programme (Bechta *et al*., 2001) is underway to examine such behaviour of fission product release from metallic and oxidic melts. The experiments will provide more understanding of species chemistry during the late phase of an accident. There are also tests aimed at examining the long-term behaviour of previously liquefied melt with an overlying water pool.

Metallic melt experiments are providing a better understanding of the important mechanisms affecting fission product and core materials releases up to 2000°C. They cover the effects of temperature, oxygen potential, sparging, slug formation, two phase pools and composition of melt.

Oxidic melt experiments are in progress, concerned with studying the volatilisation of uranium oxides, fission products and boron oxide from melts of different compositions of $UO_2/ZrO_2/SiO_2/FeO_x$. These experiments are for both air and inert atmospheres, and with different temperatures of corium.

The main chemistry interests of the work concern tellurium, ruthenium, barium and strontium and the influence of steam on the volatility of the refractory fission products and actinides. The experiments focus on the generation of these elements at both high temperature, 1000°C and low temperature 25°C.

Immersed core experiments are being carried out to determine the leaching and suspension rates of solid melts immersed in water under prototypical accident conditions. These use prototypical materials composed of UO_2 and ZrO_2 and other oxides.

15.6.5 Hydrogen and the Containment

Integral large-scale experiments on hydrogen combustion are being carried out within the EC 5th Framework programme (Bielert *et al.*, 2001) to promote understanding of the phenomena and for the development of analysis methods. These extend work carried out in the 4th Framework programme, which concluded that local information on the properties of reactive flow fields was necessary for modelling, and also that flame acceleration was important for transition from deflagration-to-detonation (DDT).

In the recent project, two different geometric scales are considered with the emphasis on different combustion regimes from slow to fast turbulent deflagrations.

Medium-scale tests with hydrogen–air mixtures were carried out within the DRIVER and TORPEDO facilities (Bielert *et al.*, 2001), i.e. in simple geometric configurations. The objectives were to study the effects on: location of the ignition, changes in blockage ratio and channel cross-section, and venting.

Large-scale tests were conducted in the RUT facility, which aimed to study the processes of turbulent flame propagation in multi-compartment geometry and in non-uniform mixtures. This facility was of a scale commensurate with typical reactor length scales. Previous tests were mainly concerned with determining critical conditions for DDT (Breitung *et al.*, 2000; Sidorov and Dorofeev, 1998). The present tests consider processes with lower flame speeds in both slow and fast deflagration situations.

To date, the data from these tests indicate specific effects of scale, multi-compartment geometry, mixture gradient effects and venting. The critical conditions for fast combustion regimes can be influenced by flow geometry, although a converging flow geometry does not especially promote this effect.

With regard to modelling, it has been found that lumped parameter codes do better at predicting slow flames, the computational fluid dynamics (CFD) codes do better at predicting fast deflagrations. There were, however, other phenomena which could not be adequately simulated by the codes, such as global quenching and pulsating flames.

Other work in the 5th Framework is aimed at investigating the effects of passive autocatalytic recombiners. The design and placing of these systems rely on a validated methodology for good prediction of hydrogen distribution under accident conditions.

15.7. SHIELDING AND CRITICALITY

The NEA also maintains a shielding and criticality database for the validation and benchmarking of methodologies for modelling different nuclear systems. For example, the database includes data from the integral shielding experiments (SINBAD). New editions of the database are being continually released. With regard to advanced technologies, radiation shielding for accelerator facilities has been the study of a recent workshop hosted by the Stanford Accelerator Centre, (NEA Annual Report, 2002). For criticality, the International Criticality Safety Benchmark Evaluation Project (ICSBEP) database, is also being developed and contains several thousands of benchmark specifications for critical or near-critical configurations.

15.8. RADIOLOGICAL PROTECTION RESEARCH

The EC research programme has included research on the understanding of radiation mechanisms to provide greater understanding of the physical, chemical, molecular and cellular biological processes as a consequence of radiation. It has also included epidemiological studies of people exposed to radiation. Collectively the mechanistic and epidemiological studies provide a good basis for quantifying the risks from radiation at low doses.

Research tasks of the former kind included the modelling of radiation ontogenesis and related biological effects and the repair of and recovery from DNA damage. Also carried out have been radiation sensitivity and molecular studies of radiation ontogenesis and predisposition to cancer and the effects of *in utero* radiation. The epidemiology of populations exposed to radiation has covered further analysis of populations exposed to large doses, e.g. atom bomb survivors through to less extreme exposures, e.g. uranium miners and radiation workers, together with the follow-up of cancers incidence. These have been complemented by further research of the treatment of exposed individuals. Other studies have addressed hereditary and genetic factors and the epidemiology of medically treated patients.

In order to evaluate radiation risks, it is necessary to have available high-quality methods for the assessment of levels of exposure to external and internal radiation. European studies have addressed the parameters that determine the fluxes of radionuclides in various ecosystems, in particular the fluxes of radionuclides in surface and groundwaters and the consequences of accidental contamination of environments. Also studied have been the intake of radionuclides and their dosimetry and the monitoring of external radiation. Code developments have been directed towards quantifying the predictions of probabilistic accident sequence codes and the development of decision supporting systems for emergency site management. Finally risk perception and communication has been studied, including comparative risk assessments of different systems.

The reduction of exposures in accordance with the ALARA principle is the primary goal of radiological protection. The main focus of research is to optimise radiological protection in many complex situations giving rise to radiation exposures, from nuclear reactor operations to participation in various other activities. Research has considered management strategies and techniques for the restoration of contaminated sites and optimisation of radiation protection of patients undergoing diagnostic radiology.

Research has also been conducted in understanding events from the past. The aims of this work are to improve the management of land (territories) that have been contaminated with radioactive material and to contribute to the future health and well being of populations that have been exposed. European research has considered in particular the consequences of the Chernobyl accident and other radiation incidents. An objective of this work is to develop more effective means for managing the radiological consequences of any future accident.

15.9. WASTE MANAGEMENT AND DECOMMISSIONING

Within the NEA joint and collaborative project programme, activities are in progress to support safety assessments of geological repositories with respect to some specific technical issues. For example, the Sorption project (NEA Annual Report, 2002) is comparing different thermodynamic modelling approaches against measured data. The thermochemical database (TDB) is being extended to include chemical thermodynamic data for the safety assessment of waste repositories. There is also the Co-operative Programme on Decommissioning and Dismantling which aims to foster information exchange between its members.

EVOLUTIONARY REACTORS

15.10. PASSIVE HEAT REMOVAL SYSTEMS

There are some phenomena that occur in evolutionary reactors under accident conditions that assume greater significance compared with presently operating plant. These mainly

Table 15.6. Passive heat removal

Issues	Experimental programmes
Primary circuit	APEX, PACTEL,
Containment and integral effects	PASCO, PANDA

Squarer *et al*. (1988), Venne *et al*. (1992), Lillington and Kimber (1997), Addabbo *et al*. (2001), Bacchiani *et al*. (1994), Kervinen *et al*. (1990), Erbacher *et al*. (1995), Wichers *et al*. (to be published) and Coddington *et al*. (1993).

relate to passive system performance, including natural circulation and passive injection, and also decay heat removal from large water pools. A major experimental programme to investigate these phenomena was carried out by Westinghouse leading up to the design certification of AP600 (Squarer *et al*., 1988; Venne *et al*., 1992) to confirm the conceptual design. Other recent programmes are shown in Table 15.6.

15.10.1 Primary Circuit Tests

Separate effects tests were carried out for the AP600 design to demonstrate the feasibility of using a passive core cooling system to mitigate all design basis accidents. There were also confirmatory tests to verify the performance of the various system components. These included: passive residual heat exchanger, automatic depressurisation, passive core cooling system check valve and core make-up tank tests.

In addition to separate effects tests, there were also passive core cooling system tests to demonstrate the overall system performance for both pressurised and de-pressurised conditions. The test facility for this programme was the Oregon State University APEX facility, and the programme was carried out within a Westinghouse/USDOE collaboration.

There were a number of thermal–hydraulic facilities commissioned and operated during the 1980s and 90s in support of the needs of currently operating plant. Many of these facilities have been dismantled but others remain either in standby or in operation to service the needs of evolutionary water reactors. Facilities include PKL, SPES for PWR, PIPER-ONE for BWR, PACTEL and PMK for VVER and PANDA for BWR (Addabbo *et al*., 2001).

The SPES facility (Bacchiani *et al*., 1994) at the SIET facilities in Piacenza, Italy was modified to include a passive core cooling system and used for high-pressure system loop thermal–hydraulic tests in support of AP600. All the safety systems were simulated and a series of tests addressed LOCA, steam generator tube rupture (SGTR) and SLB thermal–hydraulic issues. PKL is currently in use to simulate boron mixing effects, in connection with a present day reactor transient issue involving boron dilution during reflux condensation in a LOCA.

Although configured for VVER geometry, PACTEL tests (Kervinen *et al*., 1990), have been carried out to simulate passive injection during a LOCA, which is of relevance to the AP600 safety system function.

15.10.2 Containment Tests

Many of the confirmatory tests for AP600 were in justifying the passive containment cooling system. Separate-effects tests to characterise the decay heat removal characteristics of the containment design were carried out. These tests included the investigation of heat removal from wetted steel plates simulating the containment surface. Also containment external cooling air flow path pressure drop tests were carried out to characterise the frictional losses. Steam condensation tests on surfaces at different angles were performed to simulate condensation inside the containment in the presence on non-condensable gases.

Composite containments, including a steel inner liner and an outer concrete shell, have been considered to meet potential European requirements for licensing. The outer concrete shell provides greater strength to mitigate the consequences of some severe accidents. Experiments to establish passive containment cooling for such containments were carried out in the PASCO facility at FZK, Germany (Erbacher *et al.*, 1995).

Passive systems are a feature of a number of advanced evolutionary LWRs, both for primary coolant system heat removal and for containment cooling. Tests are in progress in the PANDA facility in Switzerland in the EC TEMPEST programme (Wichers *et al.*, to be published), to resolve outstanding issues of the effects of light gases for confirming the long-term LOCA response of the passive containment cooling systems for SWR100 and ESBWR.

15.10.3 Integral Effects Tests

Integral passive containment cooling tests were performed for AP600 to examine the overall containment performance at large scale. At the time, there were no other water distribution tests to provide a demonstration of water distribution over the steel containment dome outer surface and the top of the containment side walls. Wind tunnel tests were conducted to confirm the structural performance of the containment shield building air inlet and outlet.

A large-scale integral system behaviour test facility PANDA (Coddington *et al.*, 1993), is present at the PSI in Switzerland. This was originally built to understand better, long-term decay heat removal by natural circulation in passive boiling water reactors. However, since the latter is a generic phenomenon, many of the data from many of the tests are of relevance to more general light water reactor applications.

The LINX facility (Coddington *et al.*, 1993), is another facility at PSI that was used to investigate the thermal–hydraulics of natural convection and mixing in pools and large water volumes. In the past, aerosol transport was studied in the AIDA facility. This is a separate-effects facility for the investigation of aerosol transport and the associated deposition in plena and tubes.

A European Thematic network has been established for the Consolidation of the Integral System Test Experimental Databases for Reactor Thermal-Hydraulic Safety

Analysis (CERTA-TN) (FISA 2003, to be published). The objective is to preserve for the future, the reactor safety thermal–hydraulic databases acquired in various integral system test facilities. A database will be produced that has up-to-date data access and retrieval capabilities and uses modern web-based information technologies.

In the final part of this chapter some of the research requirements for future innovative reactors are addressed. Some of these also relate to work that will be needed to realise nearer term evolutionary and prototype reactor systems that will also be required to confirm the technologies of the longer term Generation IV reactors before they are built.

INNOVATIVE REACTORS

15.11. FUTURE REACTOR RESEARCH

Research programmes for the innovative designs described in Chapter 12 are described in IEA/OECD (NEA)/IAEA (2002) and Background Report for the Three-Agency Study (2001). Compared with the level of R&D investment in the performance and safety optimisation of current generation reactors over the years, and in evolutionary designs, the level of investment in future generation reactors is small at the present time.

To facilitate further research, it will be advantageous to set up collaborative international R&D programmes if possible. However there are many diverse designs under consideration and collaboration will only be possible if there are common interests in a particular field or topic. There are also the issues of commercial interests and the sharing of proprietary information to be addressed.

It is suggested in IEA/OECD (NEA)/IAEA (2002) that the setting up of a comprehensive experience database may be a useful initial activity in a collaborative relationship. Reactor designers could access this database to collect information on existing experience on the advantages and disadvantages of different reactor types.

Below are sections on the areas of research that are likely to be required for future innovative reactor systems. There are programmes already in place on research of some evolutionary systems issues; these are seen as a step towards developing the later systems. The discussion in these sections focuses particularly on the designs put forward by the GIF for Generation IV systems.

In summary, there are many R&D activities that will need to be accomplished before most of the innovative systems are available. The main technical developments for the Generation IV systems are summarised in Table.15.7. Some of these R&D activities have already started, e.g. for the nearer term SCWR and HTR concepts. SCWR activities have been ongoing since 2000 in the US, Canada, Japan, South Korea and in the EU, on materials and corrosion research. For the HTR concepts there are plans for the building

Table 15.7. Generation IV technology research

Reactor type	R&D activities
Super critical water reactor (SCWR)	Materials, corrosion, heat transfer, radiolysis and water chemistry, crack growth research in GIF countries
Very high temperature reactor (VHTR)	Fuel materials and fabrication, high-temperature materials, hydrogen production technology, graphite technology
Gas cooled fast reactor (GFR)	Fuel materials and fabrication, materials for high fluence, fuel cycle technology, safety systems
Sodium fast reactor (SFR)	Fuel cycle technology, plant simplification
Lead cooled fast reactor (LFR)	Materials, corrosion research, fuel recycle technology
Molten salt reactor (MSR)	Process chemistry, plant design research

The US Generation IV Implementation Strategy (2003), Newton (2002) and Institute of Nuclear Engineers (2004).

of a Next Generation Nuclear Plant (NGNP) at Idaho in the US for R&D as a step towards the VHTR. There are also plans for an experimental technology demonstration reactor (ETDR) looking forward to the advent of GCR technology.

15.12. HIGH-TEMPERATURE MATERIALS

15.12.1 Technical Issues

A feature of many of the innovative future designs is their relatively higher outlet temperatures compared with current generation plant. The safety envelopes of many of the component materials may not extend to these temperatures in which case new materials will need to be developed and qualified. Additionally many of the coolants may erode or corrode the surrounding materials, particularly in the high-temperature environments that are exhibited (IEA/OECD (NEA)/IAEA, 2002). There will also need to be materials developments in process systems associated with the applications of innovative reactors, e.g. hydrogen generation.

The high-temperature gas reactors may have outlet temperatures that could be as high as 1500°C. These will require significant advances in high-temperature materials, alloys, ceramics and composite materials. Future water reactors, including supercritical systems, liquid lead and molten salt systems, will also require substantial material developments to withstand both corrosive and high-temperature environments (The US Generation IV Implementation Strategy, 2003).

Anticipated areas of research could include the performance of various material compositions in these environments, the development of protective coatings and research into particular materials for specific applications.

15.12.2 Component Research

The need for research into suitable materials for components that need to withstand the aggressive high temperature and corrosive environments of future generation plants has been mentioned earlier.

Key considerations are economics and how to reduce construction cost without compromising safety. Some particular areas in HTRs where cost savings could be made included the following. The heat recovery in high temperature gas designs incorporates a large tube-in-shell heat exchanger to recover helium heat discharged from the turbine, before it is recycled. The inclusion of an advanced plate type heat exchanger would result in much reduced size and, therefore cost.

Another area is the load imposed on the helium turbine bearings. Such large turbines have not been built or operated. One option might be to increase the speed of the power turbine, thus reducing its weight and size, and therefore possibly allowing the use of gas bearings.

Much work has been done of passive devices for innovative reactors. There needs to be a better understanding of the limits of the safety devices used on present day operating reactors. The issue is how to extend the existing devices to innovative reactor applications or, if necessary, how to develop new ones.

15.13. ADVANCED FUEL DESIGN AND REACTOR PHYSICS

15.13.1 Critical Reactors

Many of the innovative systems under consideration have the capabilities to use advanced fuel cycles and there is a need for fuel performance research for many of these fuels (The US Generation IV Implementation Strategy, 2003). There are, however, a number of activities currently in progress (Table 15.8).

Innovative water reactor fuel cycle options are being considered whereby spent PWR fuel can be used for CANDUs, i.e. the DUPIC technology. This is attractive to avoid the separation of fissile material, particularly plutonium, during fuel cycle operations.

Table 15.8. Advanced fuel and reactor physics research

Issues	Experimental programmes
Advanced LWR/HWR fuel cycle	DUPIC
HTR fuel/reactor physics	OECD-NCS, IAEA CRP 5
Innovative fuels, e.g. nitride	EC CONFIRM
Plutonium burning and waste incineration	CAPRA/CADRA
Innovative fuels for ADS	EC FUTURE

The US Generation IV Implementation Strategy (2003) and Newton (2002).

High-temperature reactor fuel design is also attracting research, e.g. on the stability of particulate fuel at very high temperatures and particularly under accident conditions. There is also relatively little experience on fuel fabrication. Regarding current research programmes, there is an OECD co-ordinated research programme on the physics of plutonium/innovative fuel cycles for pebble bed reactors. The IAEA is also co-ordinating analysis of experimental results for a number of high-temperature test reactors including, HTTR (Japan), HTR (China), GT-MHR (US and Russia) and ASTRA (Russia) (Newton, 2002).

Regarding innovative fuels, nitride fuels have been considered instead of oxide fuels, because they result in lower fuel temperatures, due to their improved thermal conductivity. A wide range of different fuels has been considered for minor actinide target fuels. There are several EC research programmes on advanced fuel-related issues. The CONFIRM programme is an experimental investigation of the high-temperature stability of actinide fuel in nitride form.

There are a number of initiatives in connection with fast reactor core physics, e.g. the CAPRA/CADRA programme, which includes studies of fast reactors, particularly in connection with plutonium burning and waste incineration fuel cycles. Uncertainties in fast reactor performance are the subject of a present IAEA research programme.

Thorium fuels have been considered as an alternative to uranium for some fuel cycles and some reactor types, but there is limited experience, certainly for commercial reactors. There is limited experience in general for the molten lead cooled and molten salt cooled systems.

15.13.1.1 Accelerator Driven Systems. In some respects, there are similarities between the important issues concerning subcritical ADSs and the critical reactors. However, the application of accelerators to different subcritical systems does require some new areas of research. There is also the issue of research on accelerator systems *per se*.

There has been research into the development of fuels for ADS. The EC FUTURE programme is investigating a number of innovative oxide compounds, in solid solution and inert matrix form. The ADS is also considered in the CAPRA/CADRA programme.

There is emphasis in designing new innovative reactors (critical and subcritical) in a way to reduce the radioactive waste burden compared with existing reactors. This is being achieved through design for high fuel burn-up; the utilisation of thorium could also achieve reduction in the higher actinides produced.

A feature of many of the new designs is that new coolants are being proposed for which little operational experience exists. These coolants are considered in Section 15.14.

15.14. ADVANCED COOLANTS

15.14.1 Present Experience

The future supercritical water concepts will operate with supercritical water on either the secondary side or also possibly on the primary side. It follows that the primary to secondary heat exchangers will require special attention. The performance of conventional PWR steam generators has not been without problems and the materials used, system chemistry control and the construction methods for the supercritical systems will need significant development research. For the supercritical heavy water concepts such as CANDU X, lower cost techniques will have to be developed for separating deuterium from hydrogen in light water.

There is some relevant experience of supercritical systems from coal-fired plants. However, there is no previous experience on the use of supercritical water in high radiation backgrounds. There is also the issue of system performance under fault conditions, e.g. LOCAs, given the very high system pressures.

For the liquid lead coolant systems, there is little experience in nuclear reactors outside of Russia. There will need to be significant effort to developing the chemistry specifications and control to ensure economic and reliable performance. Lead-induced stress corrosion cracking could also be an issue.

Molten salt systems will require developments on the control of their chemistry and the coolant composition during their extended periods of operation. The high-temperature performance of key components such as heat exchangers will need to be verified. There needs to be developed isotope separation technologies to separate out the lithium isotope ^{7}Li from the naturally occurring ^{6}Li.

15.14.1.1 Natural Circulation. Existing operating water reactors rely on natural circulation to remove decay heat when forced convection is lost. Many water and heavy water cooled designs include natural convection to remove decay heat after shutdown. Some of the simpler low-pressure water reactors rely on natural circulation to remove heat at all power levels. In general, there has not been a total reliance on natural circulation in the pressurised PWR and BWR systems. The innovative liquid lead and molten salt systems also allow for some level of natural circulation, namely the removal of decay heat after shutdown. Some of the ADS systems allow for natural circulation removal of heat at all power levels.

It follows that natural circulation will be an important phenomenon in innovative reactor technology. Regarding the current state of knowledge, there has already been much work on natural circulation in current and evolutionary plant. There is greater confidence in single-phase system performance, e.g. in the gas and liquid lead systems, than in the water systems, where two-phase flow can develop. Under accident conditions the presence of hydrogen can also be a problem. Additionally, there is a need for the development of correlations for transient heat transfer under all operating conditions.

There are several projects within the EC framework research programme for evolutionary systems; these were mentioned earlier, e.g. EUROFASTNET, ECORA and FLOMIX-R. Most of these are relevant to improving the understanding of natural circulation in the evolutionary and some of the innovative reactor systems. The above programmes also cover theoretical R&D, e.g. the development of appropriate numerical methods development for CFD modelling.

15.15. PLANT OPERATIONAL RESEARCH

15.15.1 Construction

Much has been learnt from constructing the present generation of plants. Clearly, the minimising of time from the start of construction to commissioning is important. Research into finding generic ways of delivery of components to site is important.

There is a move in the evolutionary and innovative designs towards increased modularisation. Simplified designs and composite construction can reduce the amount of site work required, and therefore reduce cost.

There are some novel sitings for certain reactor types that are being proposed, e.g. barge type systems, which may be appropriate for some remote areas.

There may also be lessons that can be learned from other industries.

15.15.2 Inspection, Maintenance, Monitoring and Control

The innovative designs pose new challenges for inspection and maintenance teams. Integral systems will be very compact; there will need to be non-intrusive monitoring techniques, continuous monitoring in confined space systems. The high temperatures will require more remote and possibly robotic systems.

There has been significant progress in advanced monitoring and control. The lessons learned for the current and evolutionary systems will provide valuable input into future innovative reactor design.

15.15.3 Safeguards and Proliferation

Safeguards technologies will need to be developed for the evolutionary systems to meet the requirements of the modern world. Advanced monitoring techniques on fuel cycle routes/operations will aid these developments. There is also the issue of maintaining effectiveness of safeguards at reasonable cost, particularly for the smaller reactors.

REFERENCES

Addabbo, C., Annunziato, A., Aksan, N., D'Auria, F., Dumont, D., Nilsson, L., Rigamonti, M., Riikonen, V., Steinhoff, F., Toth, I. & Umminger, K. (2001) European network, for the consolidation of the integral system, test experimental, databases for reactor thermal–hydraulic safety analysis, EUR 20281, November 2001.

Adroguer, B. *et al*. (1999) Corium interactions and thermochemistry, CIT Project, FISA 99, EUR 19532 EN, Lux., November 1999.

Adroguer, B., Chatelard, P., Van Dorsselaere, J.P., Duriez, C., Cocuaud, N., Bellenfant, L., Bottomly, D., Vrtilkova, V., Mueller, K., Hering, W., Homann, C., Krauss, W., Miassoedov, A., Steinbruck, M., Hozer, Z., Bandini, G., Birchley, J., Berlepsch, T.V., Buck, M., Benitez, J.A.F., Virtanen, E., Marguet, S., Azarian, G., Plank, H., Veshchunov, M. & Zvonarev, Y. (2001) Core loss during a severe accident (COLOSS), EUR 20281, November 2001.

Bacchiani, M., Medrich, C., Rigamonti, M., Vescovi, O. & Conway, L.E. (1994) SPES-2, the full-height, full pressure integral system AP600 test facility.

Background Report for the Three-Agency Study. Volume 1: Innovative Reactor Features, Characteristics and R&D, 2001, http://www.nea/fr.

Bassette, D.E. (2000) Initial and boundary conditions to LOCA analysis – an examination of requirements of Appendix K, 10th International Conference on Nuclear Energy, Baltimore, MD, April 2000.

Bechta, S., Bensen, C.G., Berlepsch, T.V., Funke, F., Kissane, M.P., Koch, M.K., Kronenberg, J., Kujal, B., Manenc, H. & Newland, M.S., (2001) Late phase source term phenomena (LPP), EUR 20281, November 2001.

Benson, C.G., Bechta, S., Bowsher, B.R., Hein, H., Hellmann, S., Koch, M.K., Kortz, Ch., Kissane, M.P., Manenc, H., Mason, P.K., Kemp, C.C., Newland, M.S. & Punni, J.S.(1999) Fission Product Release from Molten Pools, Final Report, AEAT-5893 (ST: MP(99)-P32).

Bezdikian, G., Moinereau, D., Lidbury, D., Chapuliot, S., Emond, D., Streibig, L., Kerkhof, K., Budden, P., Ensel, C., Keim, E., Taylor, N., Nagel, G., Bass, R., Siegele, D., Marini, B., Sharples, J., Dahl, A. & Wadier, Y. Structural margin improvements in aged embrittled RPV with load history effects (SMILE), to be published.

Bielert, U., Breitung, W., Burgeth, B., Kotchourko, A., Scholtyssek, W., Pailhories, P., Petit, M., Studer, E., Eyink, J., Movahed, M., Petzold, K.G., Heitsch, M., Alekseev, V., Dorofeev, S., Efimenko, A., Kuznetsov, M., Okun, M., Yankin, Y. & Huld, T. (2001) Integral large scale experiments on hydrogen combustion for severe accident code validation (HYCOM), EUR 20281, November 2001.

Breitung, W., Chan, C., Dorofeev, S., Eder, A., Gelfand, B., Heitch, M., Klein, R., Malliakos, A., Shepherd, J., Studer, E. & Thibault, P. (2000) Flame acceleration and deflagration to detonation transition in nuclear safety, SOAR. NEA/CSNI/R7(2000)7, August 2000.

Coddington, P., *et al*. (1993) ALPHA, the long term passive decay heat removal and aerosol retention programme, IAEA-TC-633, IAEA Meeting, PSI, Switzerland, 25–27 May, 1993.

Cognet, G. *et al*. (1999) Corium spreading and coolability (CSC), *Proceedings of the FISA 99 Symposium, Luxembourg, 1999*.

Crouch, R., Bicanic, N., Pijaudier-Cabot, G., Bittnar, Z., Schrefler, B., McNulty, T., Smith, L. & Ciampoli, M. Modelling ageing in nuclear power plant structures: MAECENAS, to be published.

Delnondedieu, M., Coste, J.F., Kuivjes, K., Hegeman, H., Kauppinen, P., Sarkimo, M., Bros, J., Bogede Andersen, L., Dobmann, G., Acosta, B., Rogerson, A., Perosanz, F.J., Hahn, P.A., Guillemot, F., Stegermann, D., Komber, T., Niffenegger, M., Brumovsky, M., Oettel, R. & Zimmer, R. (2001) Project GRETE – evaluation on non-destructive testing techniques for monitoring of material degradation, EUR 20281, November 2001.

English, C., Sevini, F., Frund, J.-M., Langer, R., Cowan, J., Gillemot, F., Brumovsky, M., Nenonen, P., Ballesteros, A., Schmauder, S. & Bacon, D. (2001) Phosphorus influence on steel ageing (PISA), EUR 20281, November 2001.

English, C.A., Ortner, S.R., Server, W.L. & Rosinski, S.T. (2002) Review of phosphorus segregation and intergrannular embrittlement in reactor pressure vessel steels, effects of radiation on materials: *20th International Symposium, ASTM, STP 1405*, Eds. Rosinski, S.T., Grossbeck, M.L. & Kumar, A.S., American Society for Testing and Materials, West Conshohocken, PA, p. 151.

Erbacher, F.J., Cheng, X. & Neitzel, H.J. (1995) Passive decay heat removal by natural air convection after severe accidents, NURETH-7, Saratoga Springs, NY, US, September 10–15, 1995.

EUR 19843 (2001) Safe management of NPP ageing in the European Union.

Faidy, C. *et al*. (2000) Structural integrity of bi-metallic components programme (BIMET): fracture testing of bi-metallic welds, *Proceedings of ICONE 8*.

Faidy, C., Martin, G., Chapuliot, S., Kayser, Y., Safa, N., Cipierre, M.F., Gilles, P., Keinanen, H., Laukkanen, A., Sherry, A., Lidbury, D., Wintle, J., Taylor, N., Youtsos, A. & Lenkey, G. Assessment of aged piping dissimilar metal weld integrity (ADIMEW), to be published.

FISA 2001 (2001) EU research in reactor safety, EUR 20281, November.

FISA 2003 (2003) EU research in reactor safety.

Fuel Safety Criteria Technical Review, NEA/CSNI/R(99)25, July 2000.

Fujishiro, T. & Ishijima, K. (1994) NSRR experiments to study the effects of burn-up on the fuel behaviour under reactivity initiated accident conditions, 22nd Water Reactor Safety Meeting, NRC, October 1994.

Fuketa, T. (1999) JAERI research on fuel rod behaviour during accident conditions, 27th Water Reactor Safety Meeting, NRC, October 1999.

Gerard, R., Sevini, F. & English, C. (2001) The AMES strategy to satisfy European industry needs in ageing materials evaluations and studies: main achievements and prospects, EUR 20281, November 2001.

Giot, M., Prasser, H.M., Dudlik, A., Ezsol, G., Habip, M., Lemonnier, H., Tiselj, I., Castrillo, F., Van Hove, W., Perezagua, R. & Potapov, S. (2001) Two-phase flow water hammer transients and induced loads on materials and structures of nuclear power plants (WAHALOADS), EUR 20281, November 2001.

Govaerts, P. (2001) Interaction between the European Commission and National Regulatory Driven RTD Activities, EUR 20281, November 2001.

Grigoroev, V., Josefsson, B., Lind, A. & Rosborg, B. (1995) A pin-loading tension test for evaluation of thin-walled tubular materials. *Scr. Metall. Mater.*, **33** (1), 109–114.

Grigoroev, V., Josefsson, B., Rosborg, B. & Bai, J. (1997) A novel fracture toughness testing method for irradiated tubing – experimental results and 3D numerical evaluation, *Transactions of the 14th International Conference on Structural Materials (SmiRT 14), Lyon, France, August 17–22, 1997*.

Holmstrom, H., Toth, I., Prasser, H.M., Kantee, H., Elter, J., Purhonen, H., Sabotinov, L., Macek, J., Kvizda, B., Matejovic, P. & Kolev, N. Improved accident management of VVER nuclear power plants (IMPAM-VVER), to be published.

IAEA (1999), Basic safety principles for nuclear power plants, 75-INSAG-3, Rev. 1, INSAG 12, Vienna.

IEA/OECD(NEA)/IAEA (2002) Innovative nuclear reactor development: opportunities for international co-operation, IEA/OECD (NEA)/IAEA ‘Three Agency Study’.

Institute of Nuclear Engineers (2004) Generation IV Seminar on Nuclear Energy Systems Research and Development, Institute of Nuclear Engineers, DTI, London, March 2004.

Jorge, O. & Chaumont, B. (2001) French regulatory decision process for severe accident management and orientations for future European research, EUR 20281, November 2001.

Jovall, O., Larsson, J.A., Shaw, P., Klinghoffer, O., Touret, J.P. & Karlberg, G. Concrete containment management using the finite element technique combined with *in-situ* non-destructive testing of conformity with respect to design and construction quality, (CONMOD), to be published.

Kervinen, T., Riikonen, V. & Kouhia, J. (1990) PACTEL, facility for small and medium break LOCA experiments, *Proceedings of ENC '90 Conference, European Nuclear Society, Lyon, France, September 23–28, 1990*.

Krugmann, U. (2001) Design of severe accident management systems for current and future reactors: methodology, tools and research needs, EUR 20281, November 2001.

Leggart, R. *et al*. (1999) Variation of residual stresses in aged components, *Proceedings of the Conference on FISA 1999*, EC EN 19532.

Lidbury, D. *et al*. (2001) VOCALIST – an international programme for the validation of constraint based methodology in structural integrity, *Proceedings of ICONE 9*, paper 722.

Lillington, J.N. & Kimber, G.R. (1997) Passive decay heat removal in advanced nuclear reactors. *J. Hydraulic Res.*, **35** (6).

NEA Annual Report (2002), 31st Annual Report of the OECD Nuclear Energy Agency, Le Seine Saint-Germain – 12, boulevard des iles, F-92130 Issy-les Moulineaux, France.

Newton, T.D. (2002) International Activities – Reactor Physics, ANSWERS Seminar, Serco Assurance, UK.

Nordgren, G., Laperia, J., Scott, P., van Dyck, S., Roth, A., Jenssen, A., Karjalainen-Roikonen, P. & Westermarck, H. (2001) Properties of irradiated stainless steels for predicting lifetime of nuclear power plant components, EUR 20281, November 2001.

Papin, J. & Schmitz, F. (1999) Further results and analysis of MOX fuel behaviour under reactivity accident conditions in CABRI, 27th Water Reactor Safety Meeting, NRC, October 1999.

Parrat, D., Turnbull, J.A., Harbottle, J., Vitanza, C., Gubel, P., Kysela, J., Becquet, M., Manzel, R. & Pascal, C. Future needs for material test reactors in Europe (FEUNMARR), to be published.

PHEBUS Facility, IRSN, France, 2003.

Rintamaa, R. & Taylor, N. (2001) NESC strategy to satisfy European industry needs in evaluation of structural components: main achievements and prospects, EUR 20281, November 2001.

Seiler, J.-M., Sehgal, B.R., Alsmeyer, H., Kymalainen, O., Turland, B., Grange, J.-L., Fischer, M., Azarian, G., Burger, M. & Cirauqui, C.J. (2001) European Group for Analysis of Corium Recovery Concepts (EUROCORE), FISA-2001, Luxembourg, 12–14 November.

Series of AMES reports, EUR 16072 EN, EUR 16278 EN, EUR 16279 EN, EUR 16305 EN, EUR 16313 EN, EUR 16455 EN, EUR 16409 EN, EUR 16470 EN, EUR 1975 EN.

Sevini, F., Debarberis, L., von Estorff, U. & Davies, L.M. (1999) The European network on ageing materials evaluation and studies – AMES, *Proceedings of IAEA Specialist Meeting on Irradiation Embrittlement and Mitigation, Madrid, Spain, April 26–29, 1999*.

Shepherd, I. *et al*. (1999) Investigation of core degradation COBE, FISA 99, EUR 19532 EN, Lux., November 1999.

Sidorov, V.P. & Dorofeev, S.B. (1998) Influence of initial temperature, dilution, and scale on DDT conditions in hydrogen–air mixtures. *Arch. Combust.*, **18**, 87–103.

Sokolov, M.A. & Nanstad, R.K. (2000) Comparison of irradiation induced shifts of K_{JC} and Charpy impact toughness for reactor pressure vessel steels, NUREG/CR-6609, November 2000.

Squarer, D. *et al*. (1988) Test program in support of the Westinghouse advanced light water reactor, Westinghouse Electric Corporation, Research and Development Centre, 1310, Beulah Road, Pittsburgh, PA 15235, US.

Steinwarz, W., Alemberti, A., Hafner, W., Alkan, Z. & Fischer, M. (1999) Investigations on the phenomenology of ex-vessel core-melt behaviour (COMAS), *Proceedings of the FISA 99 Symposium, Luxembourg*.

Steinwarz, W., Koller, W., Hafner, W., Journeau, C., Seiler, J.M., Froment, K., Cognet, G., Goldstein, S., Fischer, M., Hellmann, S., Eddi, M., Alsmeyer, H., Allelein, H.J., Spengler, C., Sehgal, B.R., Koch, M.K., Alkan, Z., Petrov, J.B., Gaune-Escard, M., Weiss, F.P. & Bandini, G. (2001) Ex-vessel core melt stabilisation research (ECOSTAR), EUR 20281, November 2001.

The CABRI Water Loop Project, IRSN, 2002.

The US Generation IV Implementation Strategy, US Department of Energy Office of Nuclear Energy, Science and Technology, http://gen-iv.ne.doe.gov, September 2003.

Tice, D. *et al*. (1999) Evaluation of techniques for assessing corrosion cracking of dissimilar metal welds, Proceedings of the Conference on FISA 1999, EC, EN 19532.

Valo, M., Lucon, E., Brumovsky, M., Debarberis, L., Beers, M. & Kohopaa, J. (2001) Fracture mechanics based embrittlement, FRAME, EUR 20281, November 2001.

van Aerle, J. et al. (2000) *Fast Neutron Dosimetry of Nuclear Power Plants by Means of Scraping Samples Using the $^{93}Nb(n,n')^{93}Nb^{m}$ Reaction, ASTM STP 1398*, Eds. Williams, J.G., Vehar, D.W., Rudy, F.H. & Gilliam, D.S.M., American Society for Testing Materials, West Conshohoken, PA.

Vasa, I. Reactor neutronic investigations on the LP-0 reactor (RENION), to be published.

Venne, T.V.D. *et al*. (1992) The Westinghouse AP600 passive cooling test and analysis programme, International Conference on Design and Safety of Advanced Nuclear Power Plants, Tokyo, Japan, October 1992.

Voorbraak, W.P., Meer, K.v.d. & Seren, T. (2001) RETROSPEC, retrospective dosimetry focussed on the reaction $^{93}\mathrm{Nb(n,n')}^{93}\mathrm{Nb}^{\mathrm{m}}$, EUR 20281, November 2001.

WASH 1400 (1975) Rasmussen Report on Potential Containment Failures' Modes Under Severe Accident Conditions and Radioactive Releases into the Environment.

Weiss, F.-P., Rohde, U., Karlsson, R., Hemstrom, B., Lillington, J.N., Scheuerer, M., Tuomisto, H., Toppila, T., Dury, T., Cvan, M., Muhlbauer, P., Toth, I., Elter, J. & Bezrukov, Y. Fluid mixing and flow distribution in the reactor circuit (FLOMIX-R), to be published.

Wichers, V.A., Malo, J.Y., Starflinger, J., Heitsch, M., Preusser, G., Tuomainen, M. & Huggengerger, M. Testing and enhanced modelling of passive evolutionary systems technology for containment cooling, to be published.

Wiesenack, W. (1997) Review of the Halden reactor project high burn-up fuel data that can be used in safety analyses. *Nucl. Engng Des.*, **172**, 83–92.

Youtsos, T., Ohms, C., Castano, M.L., Roth, A., Gavillet, D., van Dyck, S. & van Thoor, K. (2001) Irradiation effects on the evolution of the microstructure, properties, and residual stresses in the heat affected zone of stainless steel welds, EUR 20281, November 2001.

Chapter 16

Analytical Methods Development

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Chapter 16

Analytical Methods Development

16.1. INTRODUCTION/OBJECTIVES

In parallel with current experimental research programmes, there are many activities devoted to improved model and associated computer code developments. Prior to plant application, these are validated against experimental data usually enveloping their application. With the increasing cost of experimental research, the improvement of models and the advent of faster and faster computers, the proportion of theoretical work in comparison to experimental work is increasing.

Computer codes are required for both design substantiation and safety analysis. The emphasis in this chapter will be mainly on the methods that have been developed and are available for safety analysis. A wide range of methods and codes have been developed and validated for current generation plant. This chapter examines the present status of research for current generation plant together with the implications for advanced applications. Current research is not only targeted towards more advanced modes of operation of existing plant but also for application to advanced, particularly evolutionary, plant applications. In many instances, code validation for current plant remains valid for evolutionary plant. This chapter does, however, describe how new phenomena relevant to evolutionary plant can be modelled, e.g. associated with passive system performance.

This chapter considers the role of different types of codes, integral, system, lumped parameter, computational fluid dynamics (CFD) and other specialist codes in the context of reactor design and safety research. More stringent safety standards imply more exacting quality assurance standards for all levels of code development, verification, validation and applications. Advanced software techniques offer more automated tools. Modern computer platforms enable detailed safety analyses to be performed that were not feasible at the times of licensing of many of today’s plants. These theoretical topics are covered.

The primary focus of this chapter will be on water reactor technologies. Water reactors occupy the overwhelmingly largest fraction of existing reactors in operation today and the nearest term evolutionary reactors are also likely to be of this type.

Analytical methods have been developed for other reactor types and some of these will be developed further as the innovative gas and liquid metal reactors’ concepts move forward. They will receive a brief mention in the last section of this chapter on innovative reactors. Analytical methods developments for the innovative reactors will proceed in parallel or slightly behind the corresponding experimental programmes that will be needed to develop innovative reactor technology. The latter were described in the previous section.

CURRENT GENERATION REACTORS

16.2. PLANT LIFE EXTENSION

Computer codes are being developed by FISA 2003: EU Research in Reactor Safety (to be published) and Van Duysen *et al.* (2004a) to assess the irradiation effects on materials in relation to plant life extension of existing LWRs. These are being termed "Virtual Reactors". Multi-scale modelling codes are being developed that can simulate at various scales, e.g. atomic scale (nanometres) up to mesoscale (micrometres) and macroscopic scale (centimetres). The aim is to develop codes that can predict the response of materials to any realistic situation especially in conditions that are difficult experimentally.

Examples of EC projects include the following activities (Table 16.1).

The REVE project (Jumel *et al.*, 2000) aims at simulating irradiation effects in RPV steels of LWRs. The project has built the virtual reactor RPV-1. It has parameters that simulate the key in experimental programmes that are found to be important, e.g. irradiation (neutron spectrum, temperature) and tensile properties (deformation rate) and material properties.

Another project is the SIRENA project (Jumel *et al.*, to be published) that aims at extending the REVE project for pressure vessel steel to Zr–Nb fuel assembly cladding. It will eventually consider the stress-corrosion cracking behaviour of these irradiated alloys in an iodine-rich environment.

ITEM (Van Duysen *et al.*, 2004b) will act as a vehicle for sharing experience of different users on multi-scale simulation. It will provide a validation data-base for code users.

It has been observed in practice that the fracture toughness measured in test specimens is less than that exhibited by cracks in components and current methodologies therefore underestimate failure margins. VOCALIST (Lidbury *et al.*, to be published) has the objective of developing models to allow for the constraint effect in predicting the component fracture behaviour.

There are future initiatives within the prediction of irradiation effects on nuclear reactors components (PERFECT) project to extend the RPV-1 methodology and to build new virtual reactors to simulate irradiation effects on stainless steels.

Table 16.1. Plant life extension

Phenomena simulated	EC research programme (computer code)
Irradiation effects in RPV steels of LWRs	REVE (Virtual reactor (RPV-1))
Irradiation of Zr–Nb fuel assembly cladding	SIRENA
Multi-scale simulation	ITEM
Component fracture behaviour	VOCALIST
Irradiation effects on SS	PERFECT (Extension of RPV-1)
Ageing of concrete vessels	SIFEL, CONMOD

FISA 2003: EU Research in Reactor Safety (2003).

Finite Element codes are being developed for structural analysis applications. Examples are the SIFEL code, applied to the modelling of ageing of concrete vessels (Crouch *et al.*, to be published) and also the CONMOD code (Jovall *et al.*, to be published), a finite element technique for the modelling of concrete containments.

CURRENT GENERATION AND EVOLUTIONARY REACTORS

16.3. FUEL BEHAVIOUR

Many analytical methods have been developed for studying the behaviour of LWR fuel for safety analysis. Steady-state codes are used to define initial conditions for transient analysis. Transient codes are used to show compliance against acceptance criteria. The validity of the fuel models is constantly under review as fuel rods are being extended to higher burn-up. Also different clads are under consideration that may respond differently to fuel behaviour, e.g. swelling under accident conditions.

16.3.1 Steady-State Fuel Performance

Many codes incorporate single rod models, which calculate thermal properties such as stored energy, radial temperature profiles, fission gas release to the gap and mechanical properties such as creep deformation and irradiation growth (NEA/CSNI/R(99)25, 2000). Examples of such codes are COMETHE, FRAPCON, METEOR, TOUTATIS, TRANSURANUS and ENIGMA (Bailly *et al.*, 1999; Table 16.2). For LOCA analysis, it is important to calculate initial stored energy from normal operation conditions. Other parameters that need to be calculated are clad oxidation thickness, the internal gas pressure, and geometrical parameters including the axial clearance between rods and end fittings. It is important to calculate fission gas content in fuel grain boundaries, fuel porosities and fission gas movement between grains and grain boundaries for the analysis of fuel failure mechanisms in RIA transients. Under RIA conditions, pin failure may result if sufficient fuel swelling and grain swelling occur.

In order to calculate these properties fuel performance codes include a wide variety of models for calculating: radial power profiles, thermal conductivity and specific heats of

Table 16.2. LWR fuel performance

Phenomena	Computer code/model
Steady-state Transient (RIA & LOCA)	COMETHE, FRAPCON, METEOR, TOUTATIS, TRANSURANUS, ENIGMA FALCON/FREY, FRAPTRAN, SCANAIR

NEA/CSNI/R(99)25 (2000).

materials, gap conductance, hydrogen absorption, waterside corrosion, creep properties, mechanical properties, creep properties, stress–strain relationships, fuel densification, and fuel swelling.

In the future, these codes are likely to be called upon to model burn-ups of up to 65 MWd kgU^{-1} or higher. The FRAPCON code has recently been modified for burn-ups up to 65 MWd kgU^{-1} (Lanning *et al.*, 1997). Many of the current codes/models were originally developed and validated for more moderate burn-ups of 40 MWd kgU^{-1} and the applicability of these codes at higher burn-ups is under review. The models will also require review with regard to their application to MOX fuel.

16.3.2 Cladding Performance

Advanced clads are being developed to exhibit better corrosion, mechanical properties and reduced growth under normal operating conditions. The models are under review for application to different cladding materials. As noted above, the clads may also experience different loads from newer fuels, e.g. MOX fuels compared with more traditional uranium oxide fuel.

16.3.3 Transient Fuel Rod Codes

Transient codes have been developed that include not only the physical models of the steady-state codes but also include additional modelling for transient thermal behaviour, heat capacity and heat transfer, transient mechanical properties such as long-term creep, cladding plastic stress–strain phenomena and ballooning at high temperatures. Other effects such as the effects of annealing, oxidation and hydriding, and changes of phase will also be modelled. Examples of transient fuel rod codes are the EPRI codes, FALCON/FREY, FRAPTRAN and the French IRSN code SCANAIR (IRSN: Scientific and Technical Report, 2002).

The main purpose of the transient fuel codes is for analysing the fuel rod response for RIAs and LOCAs. The main issues in modelling are related to the time-scales of different transient phenomena in relation to the time-scales of these transients. For example, fission product release may occur on both short-term and long-term time-scales. The time-scales of non-transient swelling and axial growth are much longer than the above accident transient time-scales. Different codes include different modelling assumptions, e.g. in addition to modelling pre-failure fuel behaviour, some of the codes include rod failure models.

At higher burn-ups, for example greater than 40–50 MWd kgU^{-1}, the Rim zone in the fuel requires special modelling attention. This is to make sure that the degradation in fuel thermal conductivity caused by structural changes in the fuel in this region is correctly modelled. Further differences in modelling requirements may also exist for MOX fuel and for advanced clads compared with more traditional UO_2 fuel and clads.

Regarding other reactor types, HTR technology was under development in the 1980s but is now believed to be a realistic alternative to LWR. Ceramic fuel technology has been established but further research is required to ensure that fuel performance is sufficiently reliable at high temperature (Hesketh, 2001). Current research is focussing on the fuel manufacturing process but methods will need to be developed to demonstrate that the fuel will be reliable to its design discharge burn-up.

Modelling codes for liquid metal fast reactors have been developed in various national programmes (IAEA-TECDOC-1083, 1999). The principal codes are TRAFIC (UK), GERMINAL (France), SATURN-TRANSIENT (Germany), LIFE (US), CEDAR (Japan) and KONDOR (Russia). These codes have a reasonable validation for moderate levels of burn-up (less than 12–15 at.%). The codes predict fuel pin thermal and mechanical behaviour for oxide fuels in steady-state and transient conditions. Some of these codes, e.g. TRAFIC also describe the behaviour of fuel pins after failure.

16.4. REACTOR PHYSICS

16.4.1 Nuclear Data

The multi-dimensional reactor kinetics codes discussed below require neutronic cross-sections that are provided by nuclear data codes. Nuclear data are required that in general will vary with burn-up and other time-dependent parameters, such as fuel and moderator temperature. A large number of computer codes are available that have been developed for LWR and other reactor applications. Table 16.3 shows a sample of existing codes.

The UK code WIMS is a typical example of one such code (Halsall, 1995; Hutton, 2000). WIMS is a modular reactor physics code for neutronics calculations. It provides a range of calculational capabilities, from pin-cell modelling to whole core modelling of power and flux distributions, using diffusion theory, discrete-ordinates, collision probability, characteristics or Monte Carlo techniques.

Data from these codes are also used to provide input for 3D (static) core simulators. The typical inputs are power distributions, burn-up distributions, reactivity worth and core-wide reactivity coefficients.

The results from nuclear data codes are also used for determining thermal–physical properties in the transient codes that depend on burn-up, e.g. thermal conductivity. In conjunction with a good calculation of fuel composition, there then exists a methodology for providing an accurate prediction of decay heat.

Table 16.3. Reactor physics

Phenomena	Computer code/model
Nuclear data	WIMS
Reactor kinetics	PANTHER, RAMONA, PARCS, SIMULATE-K, CORETTRAN, SAPHYR

NEA/CSNI/R(99)25 (2000).

These codes are also being applied to cores with advanced fuels, e.g. MOX.

They are being applied to cores with high burn-up fuel. Here the main issue is concerned with determining the fuel composition, since new reactor physics at high burn-up is not expected.

16.4.2 Reactor Kinetics

Reactor kinetics codes are used to calculate assembly averaged neutron flux and power distributions in a reactor core under transient conditions. The UK PANTHER code developed by British Energy is a typical example (Hutt, 1996). It includes a neutron diffusion neutronics model, coupled with a 1D thermal hydraulics model for the core region. The code has also been coupled with the RELAP5 system thermal hydraulics code, see below, to provide a neutronic/primary circuit modelling capability. The code can perform reactor calculations, fuel management studies and safety transient analysis. It can also be used for on-line calculation support. Other codes include: RAMONA, PARCS (Joo, 1998), SIMULATE-K, CORETRAN and SAPHYR.

Whole-core events, such as macroscopic temperature changes cause global power changes and these can be modelled adequately with point-kinetics models. These models require as input, reactivity coefficients, the effective delayed neutron fraction, generation time and control rod worths. However, in some transient conditions such as rod ejection or control rod drop, localised events occur that require multi-dimensional neutron kinetics analysis with codes of the type mentioned above. One-, two- and three-dimensional models require neutronic input parameters such as assembly averaged neutron cross-sections and delayed neutron fractions that are obtained from the nuclear data codes. The neutronics codes typically model energy groups condensed into two energies. Delayed neutron fractions would usually be modelled on a nodal basis.

A review of applicable existing thermal gas cooled reactor experience and previous gas cooled fast reactor projects is given by Mitchell *et al.* (2001). Gas reactor physics methodologies have been established, e.g. WIMS and PANTHER for application in the UK gas reactor industry. Gas reactor physics methodologies are being extended to high-temperature reactor applications with pebble bed fuels, taking advantage of already existing experience. The fuel and core design for gas cooled fast reactors are necessarily different, e.g. the graphite pebble bed concept cannot be used because graphite is a moderator and also because of the fast neutron core reactivity sensitivity to geometry variation. The fast gas reactor core will probably be based on more conventional LMFBR design using MOX or UOX steel clad pellets, e.g. as in the ETGBR design.

A range of computer codes has been developed for fast reactor neutronics (IAEA-TECDOC-1083, 1999). These include codes based on classical diffusion theory, transport theory and Monte Carlo methods. Within the European fast reactor community, the European reactor analysis optimised system (ERANOS) code system has been developed. This system embodies a modular system of codes not only for performing neutronic system design calculations but also for experimental analysis of critical facilities.

Undoubtedly, more research will be needed to develop reactor physics methodologies for the evolutionary plants and certainly for the more innovative concepts. However, there already exists substantial pool of experience on which to build.

16.5. THERMAL–HYDRAULICS

Thermal–hydraulics modelling for LWRs has been largely concerned with two main issues. The first is in demonstrating that adequate safety margins to boiling exist under normal operation conditions. The second is in demonstrating the effectiveness of the safety systems in preventing core melt in various design basis accidents such as LOCAs and intact circuit transients. To this end, there has been major investment in large experimental facilities and associated code developments. Remaining research requirements are concerned with developing improved modelling of 3D and multi-phase flow conditions.

16.5.1 Sub-Channel Analysis

Sub-channel analysis is performed to determine the safety margin to boiling in peak rated channels in LWR assemblies. The flow and heat transfer distributions inside a fuel assembly can be analysed by sub-channel codes. The usual reason for analysis is to demonstrate compliance with the 'Departure from Nucleate Boiling Ratio' (or DNBR) requirements. The codes calculate the DNB from various channel-averaged parameters. Well-known sub-channel codes are COBRA and VIPRE (Table 16.4).

In such codes, two-phase flow is normally treated via a 3D flow model, which is coupled to a 1D model for fuel rods of different ratings. A detailed model of the heat transfer between the surface of the cladding and the coolant is included. The critical heat flux is calculated with a correlation.

Most fuel bundle designs are complex and it is necessary to consider the effect of such geometries on the DNBR. Modern fuel bundle designs may include part-length rods and/or large water holes and these are clearly difficult to model. Grids of varying design exist for

Table 16.4. Thermal–hydraulics

Code type	Computer code/model
Sub-channel	COBRA, VIPRE
Transient analysis system	TRAC, RELAP5, TRACE, CATHARE, ATHLET, RETRAN
CFD codes (CFMD in development)	CFX, FEAT, FLUENT, CODE-SATURNE, TRIO-U, FLUBOX

Guffee *et al.*, RELAP5/MOD3 Code Manual (1995), Spore *et al.* (2001), Page *et al.* (1998), CFX 4.3 (1999), Weiss *et al.* (to be published), Scheuerer *et al.* (to be published), Paillere *et al.* (to be published) and Yadigaroglu (to be published).

support and promote mixing. There is a need to improve sub-channel codes to take account of these features and ensure, in particular, that void distributions are adequately modelled.

New fuel vendors supply correlations for their individual fuel rod designs. These are developed for fresh fuel and generally do not include the effects of burn-up, so their adequacy for highly irradiated fuel needs to be established. In highly irradiated rods, the surface may be significantly oxidised with different thermal–hydraulic performance characteristics. A particular issue may be different boiling characteristics and any influence on critical heat flux needs to be established.

16.5.2 Transient Analysis

Large thermal–hydraulic system codes have been developed for the analysis of various fault conditions and initiating events. Examples of such codes include TRAC (Guffee *et al.*), RELAP5 (RELAP5/MOD3 Code Manual, 1995), CATHARE, ATHLET and RETRAN together with other industry system codes. Recently the TRAC-M or TRACE code is being developed which constitutes an amalgamation of the TRAC and RELAP5 codes (Spore *et al.*, 2001). For the PWR, these codes calculate the flow, temperature and pressure in the primary circuit and secondary side. They include modelling of the reactor vessel, hot and cold legs, pressuriser and steam generators and safety systems using fundamental components of pipes, vessels, valves, etc. Most of the system codes can be adapted to other water reactor systems, e.g. BWR, VVER and RBMK.

In addition to thermal–hydraulics models, these codes typically contain point kinetics models to model the reactor power, and also 1D (radial) fuel rod models. Many have now been coupled to 3D neutronics codes of the type described above. In the UK, for example RELAP5 has been coupled with the PANTHER code, e.g. using the TALINK code (Page *et al.*, 1998). RELAP5 has also been coupled with other neutronics codes. Generally, a few individual fuel rod models are coupled to a single thermal–hydraulic channel, e.g. an average rod and a hot peak rated leading rod. The fuel rod/coolant heat transfer exchange includes cladding to coolant heat transfer correlations, a gap conductance model between the fuel and clad, and thermophysical properties for the fuel. Ballooning, oxidation and rupture models are also required for the clad for LOCA analysis.

16.5.3 Computational Fluid Dynamics

Computational dynamics (CFD) codes provide solutions to modelling more general thermal–hydraulics situations, which are not modelled adequately by the system codes. Examples of such codes are the CFX code (CFX 4.3, 1999) developed for general fluid flow applications; in particular, it can be applied for reactor safety analysis. Another example is the FEAT code including coupled thermal–hydraulic and structural modelling capabilities developed by British Energy.

CFD codes are used to model flows where 3D effects and/or turbulent mixing phenomena are important. They are also useful in modelling complex geometries with arbitrary boundary shapes and internal structures. They are used for detailed phenomenological modelling to gain understanding but also in supplying mixing models for benchmarking system codes. For LWR applications, they are used in transient analysis of boron dilution events, thermal mixing in overcooling transients, and cold water mixing in steam line breaks. They are also used for modelling pools in advanced reactor passive systems where thermal mixing processes are often important in modelling heat transfer mechanisms.

CFD codes are being validated for reactor safety applications in a number of different European research projects. The codes include CFX-5 and FLUENT for modelling flow mixing and flow distribution in the primary circuit (FLOMIX-R) (Weiss *et al.*, to be published). CFX-5, CODE-SATURNE and TRIO-U (ECORA) (Scheuerer *et al.*, to be published) are being validated for a range of applications including primary loop flow mixing, pressurised thermal shock (PTS) flow modelling and 3D containment analysis.

To date, CFD codes applications in reactor safety are largely concerned with single-phase applications. The ASTAR project (Paillere *et al.*, to be published) has looked to extend the modelling limitations of the systems codes such as CATHARE, ATHLET, TRAC, RELAP5, etc. For example, a multi-dimensional model FLUBOX was coupled to ATHLET within this project. CFD codes are now being developed for multi-scale (termed CFMD (Yadigaroglu, to be published)) applications and these are being examined at the research level.

Fluid flow modelling in gas reactors where only single-phase flows are present, is a much more straightforward proposition than the modelling of two-phase flows in LWRs. CFD codes have been applied to gas reactor flow modelling in normal operation and accident conditions. They are particularly amenable for modelling such flows and they have also been coupled with neutronics codes to provide power variation feedbacks.

There have been substantial analytical methods developments for modelling sodium cooled LMFBRs (IAEA-TECDOC-1083, 1999). Codes have been developed with the support of extensive experimental facilities in Europe and the US. Codes have been produced for modelling decay heat removal under various accident conditions. The requirements have been to model forced and natural circulation in various components under steady-state and transient conditions. There has been particular attention paid to the development of multi-dimensional codes for modelling disturbed turbulent liquid metal flows. Much of this experience will be relevant to future liquid metal systems.

16.6. SEVERE ACCIDENTS

16.6.1 Integral Codes

Integral computer codes are being developed to provide a LWR accident analysis capability for modelling the course of a severe accident through its various stages.

Table 16.5. Severe accidents

Code type	Computer code/model
Integral	ASTEC, ECART, MELCOR, MAAP
Mechanistic	SCDAP/RELAP5, VICTORIA, CONTAIN

Jacq and Allelein (2000), Allelein *et al.* (2000, 2001), NUREG/CR-6119 (1998), IAEA-TECDOC-752 (1994), Allison *et al.*, NUREG/CR-5545 (1992) and NUREG/CR-6533 (1997).

They provide a phenomena coupling capability from degradation of the fuel rods through to formation of a molten pool and if the accident progresses unchecked, to the containment loading and release to the environment. They include modelling for the release of fission products and aerosols (e.g. from control rod materials and core–concrete interactions). They include models for fission product transport through the reactor coolant circuit to the containment, including deposition, re-suspension of aerosols and also the fission product source to the environment, should the containment fail or be vented by operator action.

Examples of such codes include ASTEC (Jacq and Allelein, 2000; Allelein *et al.*, 2000; Allelein *et al.*, 2001), ECART, MELCOR (NUREG/CR-6119, 1998) and MAAP (IAEA-TECDOC-752, 1994; Table 16.5). These have been validated against various severe fuel damage and fission product release experiments during the course of their development. Further data are now becoming available from the integral PHEBUS FP experiments. The first objective of PHEBUS is specifically to provide high-quality data on the strongly coupled processes that occur in severe accidents, as described above. The second objective is to validate the codes against these data and to define the envelope of validation of the codes. The PHEBUS programme is ongoing currently. In addition to integral analysis interpretation, it is supported by additional analysis from detailed CFD codes.

The Accident Source Term Evaluation Code (ASTEC) aims to model all stages of a severe accident sequence from the initiating event through to fission product release from the containment. It is a European code developed by GRS (Germany) and IRSN (France). The code adopts a best estimate approach and aims to include all the major phenomena and their interactions and also the main plant systems. The other requirements are that it should be fast running, flexible for performing sensitivity analyses and with appropriate validation. The code has been made available to the EC European Validation of the Integral Code ASTEC (EVITA) 5th Framework Project for further validation activities. The applications of the code are for determination of source terms, support to level 2 PSA and to promote better understanding of the physical phenomena.

16.6.2 Mechanistic Codes

More mechanistic codes have been developed to model in detail, the various phases of a severe accident. They include SCDAP/RELAP5 (Allison *et al.*) for the in-vessel core melt-down phase, VICTORIA (NUREG/CR-5545, 1992) for fission product effects including transport in the primary circuit and CONTAIN (NUREG/CR-6533, 1997), a containment phenomenology code.

The development of severe accident LWR codes (and supporting experimental programmes) has attracted significant research and development investment, much greater than that invested in other reactor types. The work have resulted in the development of improved accident management guidelines for the existing plants and improved robustness against severe accident challenges in the design of evolutionary plant.

16.7. CRITICALITY AND SHIELDING

Other analytical methods are available to support more general nuclear plant operation such as ex-reactor fuel store management or radiation dosage evaluation, e.g. operations that require criticality or shielding modelling capabilities. These are described in this section; representative codes are shown in Table 16.6.

Monte Carlo techniques provide the most accurate way of determining the multiplication factor k-effective for systems containing fission. The MONK code (Smith *et al.*, 2000) is one such code for determining nuclear criticality margins and safety.

Shielding codes such as MCBEND also use Monte Carlo techniques to determine radiation levels arising from nuclear sources (Wright *et al.*, 1999). RANKERN represents a particular methodology for shielding using point kernel techniques for gamma-ray transport solutions (Chucas and Curl, 1999).

MCNP (NEA Annual Report, 2002) is another Monte-Carlo Code System for radiation dosimetry modelling. MCNPX extends this capability to high energy applications.

In terms of research requirements, these codes are relatively mature. Research is taking place to improve the numerical methods in these codes, and also in regard to extending their ranges of application to other nuclear systems.

In the next section, some of the model and code research and developments that are specific to evolutionary reactor systems are reviewed. Particular attention is paid to water reactors. There are some phenomena, particularly relating to the passive evolutionary systems, which are more important or new compared with existing plant and these therefore require further attention.

Table 16.6. Criticality and shielding

Phenomena	Computer code/model
Criticality	MONK
Shielding	MCBEND, RANKERN, MCMP

Smith *et al.* (2000), Wright *et al.* (1999), Chucas and Curl (1999) and NEA Annual Report (2002).

Table 16.7. Specific modelling for advanced water reactor safety analysis

Phenomena	Computer code/model
Zircaloy oxidation	SCDAP/RELAP5, ICARE/CATHARE, ATHLET-SA, MELCOR, MAAP
Boron–carbide reactions	SCDAP/RELAP5, MELCOR, MAAP
Aqueous fission products	MELCOR, MAAP
Inerted containment atmosphere	MELCOR, MAAP, JERICO, FUMO
Oxygen ingress into inerted containment	MELCOR, MAAP, CONTAIN
Hydrogen effects on natural circulation	MELCOR, MAAP
Heat exchange to the containment shell	CONTAIN, MAAP (AP-600)
Direct containment heating	CONTAIN, MAAP

IAEA-TECDOC-752 (1994).

EVOLUTIONARY REACTORS

16.8. ADDITIONAL EVOLUTIONARY WATER REACTOR MODELLING

Many of the codes that are mature for current generation LWR plant will be applicable to evolutionary plant (Table 16.7). Further many of the anticipated developments ongoing for present generation plant, e.g. better modelling of thermal mixing, buoyancy effects in primary circuit, 3D modelling of containment, are also relevant to the modelling of evolutionary plant with greater dependence on passive safety systems. Thus, improvements in CFD and CFMD codes will be relevant. Codes have already been developed for licensing certification of reactors such as AP600 and AP1000. Improved passive containment models for lumped parameter codes WAVCO and SPECTRA are already in progress (Wichers *et al.*, to be published).

16.8.1 Passive Heat Removal Systems

There are some features of evolutionary plants that require new models and extension of the codes that have been developed for present day plants.

Some integral codes have special models that have been developed for particular plants. They therefore cannot be applied or it is difficult to apply them to new plants. It is also difficult to apply them to new experiments for the purposes of code validation. The MAAP code developed by EPRI is an integrated severe accident code, which has specific models for specific plants and phenomena. A special version AP600-MAAP has been developed for evaluation of AP600 safety (IAEA-TECDOC-752, 1994).

In general, in advanced LWR designs, there is a requirement for much stronger thermal–hydraulic coupling between the primary circuit and containment. This has led to the coupling of some system thermal–hydraulic codes, e.g. RELAP5 with the containment code CONTAIN.

Many evolutionary passive designs have large pools as heat sinks and condensers. To be effective, these need to be well mixed and the effectiveness of these pools needs to be established. The system thermal–hydraulic codes do not have the required mechanistic mixing models and therefore need to be benchmarked against CFD codes.

The system codes have limitations in their modelling of condensation, particularly in the presence of non-condensables or 3D effects.

Finally, it has been established that the performance of the system codes in buoyancy-driven situations is less robust, than in their application to the modelling of high-pressure forced convection flows, the regimes for which they were originally developed. Much effort has been expended in improving the performance of these codes in low-pressure applications in current generation reactors, e.g. in the modelling of shutdown accidents. Generally, later versions of the system codes, e.g. RELAP5 are much more robust (compared with earlier versions in this respect).

16.8.2 Structural Assessment

Finite element techniques have offered a substantial modelling improvement capability over the more classical mechanical equilibrium codes. They can be used for evaluating stresses, strains and displacement of components for different accident situations. They can model both static and dynamic effects. They can be used to evaluate the failure mode of structures, e.g. containments under increasing loadings (IAEA-TECDOC-752, 1994).

It is considered desirable (Sammarato *et al.*, 1992) that the methodologies for future advanced containment should be based on 'best estimate' approaches. The more advanced codes all offer this capability. The traditional modelling approaches were generally much more conservative.

The development of improved codes can only be realised if there are corresponding improvements in input data. There is a need therefore for ensuring that adequate materials data are available.

Dynamic load modelling under severe accident loads is now within the capabilities of the computer codes but the problem may be in specifying the appropriate boundary conditions, e.g. in assessing the load resulting from a hydrogen detonation. The thermo-mechanical assessment of core catchers is also a modelling requirement for the assessment of advanced containments. This has been investigated in France (Millard *et al.*, 1992) and Italy (De Rosa *et al.*, 1992).

INNOVATIVE REACTORS

16.9. MODELLING REQUIREMENTS FOR INNOVATIVE SYSTEMS

Modelling capabilities are increasing rapidly in many areas, alongside the development of continually improving hardware with ever-increasing capacity. Hardware developments will continue to develop enabling modellers to produce increasingly detailed computer models for design optimisation and safety analysis. Improved software will result in improved code architectures and facilitate better quality assurance.

Modelling developments can be anticipated in certain areas. While the detailed modelling of single-phase flows is sufficiently mature, this is not the case for multi-fluid dynamics. Future safety analysis will need to accommodate the modelling of such flows and new methodologies will need to be developed. There is also the need to model better the various scales of turbulence for the purposes of modelling flow mixing processes, e.g. thermal or boron mixing (in water reactors).

The licensing of future systems will require a broader design basis than that deemed acceptable in the present generation of reactors. This may require the development and validation of new methodologies to be able to perform safety analysis for this increased envelope.

For economic reasons, more realistic margins will need to be calculated to ensure that plant performance, e.g. power output is optimised. Thus conservatism in modelling will need to be reduced and there will be a need for methodologies to model more coupled phenomena, e.g. thermal–hydraulics and neutronics, or thermal–hydraulics and structural response phenomena.

Computer model developments for the innovative systems will proceed in parallel with experimental research on materials to withstand high temperature and corrosive environments, etc. This was covered in the previous chapter. In many cases, it may be possible to extend the present day models with appropriate development and validation against prototypical data.

Some work has started on research for the development of the nearer term Generation IV reactor concepts, SCWR and VHTR (Table 16.8).

In the SCWR R&D programme, plant (core, channel, vessel, containment and balance of plant (BOP)) design is in progress in some of the GIF member countries, principally in US, Canada, Japan, EU and Russia (Generation IV Seminar on Nuclear Energy Systems Research and Development, 2004). Analytical work (supported as appropriate by experiments) in progress includes research on stability analysis, materials research, corrosion, heat transfer, radiological and water chemistry, crack growth, modelling of transport phenomena, safety methods development, thermal cycle optimisation and fuel cycle analysis.

Table 16.8. Specific modelling for innovative reactors

Phenomena	Computer code/model
Materials, corrosion	Extension of existing models supported with new materials thermophysical data
Fuel cycle and reactor physics	Extension of existing water, gas and liquid nuclear models develop better transport codes as appropriate
Fluid flow/heat transfer	Extension of system and CFD codes for advanced coolants. Extend coupling capabilities with neutronics/structural models

Generation IV Seminar on Nuclear Energy Systems Research and Development (2004).

For the high-temperature gas concepts VHTR and GFR, design of the plant systems is also in progress, particularly in relation to the reactor physics, fuel technology and high-temperature materials. Safety studies are being carried out in parallel. For the GFR, research is also being performed on aqueous and pyro reprocessing.

For the liquid metal systems, there has been a considerable amount of theoretical work on liquid sodium systems, see e.g. IAEA-TECDOC-1083 (1999). Analysis methodologies have been developed; these will need to be revalidated for the new concepts being proposed. For the lead systems, the primary experience resides in Russia. Theoretical wok will be required to support the fuel cycle.

For molten salt, models will be needed for modelling basic thermodynamics of the new fluid, integrated waste treatment and nuclear data.

There will be some ADS systems modelling requirements, additional to those required for the critical reactor concepts, e.g. one area is in nuclear data model development for sub-critical cores (IAEA-TECDOC-985, 1997).

It is clear that different GIF countries have different requirements and therefore research priorities. There is also a difference in the level of commitment across the participating countries. The EC is now a GIF member in its own right and has instigated the Michel Angelo Network (MICANET) programme (Ion *et al.*, 2003) to steer the EC towards an appropriate R&D strategy that enables the nuclear option to be left open in the EU via the development of innovative systems.

REFERENCES

Allelein, H.-J., Schwinges, B. & Van Dorsselaere, J.P. (2000) Validation of the integral code ASTEC, International Meeting on Best-estimate Methods in Nuclear Instillation Safety Analysis (BE-2000), Washington D.C., US, November 2000.

Allelein, H.-J., Neu, K., Van Dorsselaere, J.P., Sdouz, G., Fischer, K., Holzbauer, H., Passalacqua, R., Jones, A.V., Muller, K., Dienstbier, J., Koch, M.K., Burger, M., Buck, M., Artnik, J., Kostka, P., Lajtha, G., Dominguez, T., Kubisova, L., Matejovic, P., Bujan, A., Slaby, J., Hering, W., Azarian, G. & Ranval, W. (2001) European Validation of the Integral Code ASTEC (EVITA), FISA: EU Research in Reactor Safety, EUR 20281, November 2001.

Allison, C.M. *et al.* (1996) SCDAP/RELAP5/MOD3.2 Code Manual, NUREG/CR-6150, INEL-96/0422.

Bailly, H., Menessier, D. & Prunier, C. (1999) *The Nuclear Fuel of Pressurised Water Reactors and Fast Neutron Reactors*, Lavoisier Publishing.

CFX 4.3 (1999) *CFX Documentation*, CFX International, AEA Technology plc, Oxfordshire, UK, 1999.

Chucas, S.J. & Curl, I.J. (1999) Streaming calculations using the Point-Kernel Code RANKERN, *Proceedings of the International Conference on Radiation Shielding ICRS'9, Tsukuba, Japan, October 1999*.

Crouch, R., Bicanic, N., Pijaudier-Cabot, G., Bittnar, Z., Schrefler, B., McNulty, T., Smith, L. & Ciampoli M. Modelling ageing in nuclear power plant structures: MAECENAS, FISA 2003: EU Research in Reactor Safety, to be published.

De Rosa, F. *et al.* (1992) ENEA activities on containment behaviour, TCM on advanced containment technologies, IAEA-TC-818, Aix-en-Provence.

FISA 2003: EU Research in Reactor Safety, 2003.

Generation IV Seminar on Nuclear Energy Systems Research and Development (2004), Institute of Nuclear Engineers, DTI, London, March 2004.

Guffee, L.A. *et al.* (1991) TRAC-PF1/MOD2 Code Manual. Programmer's Guide, NUREG/CR-5673, LA-12031-M.

Halsall, M.J. (1995) WIMS – A general purpose code for reactor core analysis, December 1995.

Hesketh, K. (2001) Advanced fuel designs for existing and future generations of reactors: driving factors from technical and economic points of view, FISA : EU Research in Reactor Safety, Luxembourg, 12–15, November 2001, EUR 20281.

Hutt P.K. (1996) PANTHER User Guide for Release 5.0, TNGF/REP/0073/93, EPD/PANTHER/UG/5.0, November 1996.

Hutton, J.L. (2000) New capabilities of the WIMS code, PHYSOR'2000, Pittsburg, US, May 2000.

IAEA-TECDOC-752 (1994) Status of advanced containment systems for next generation water reactors.

IAEA-TECDOC-985 (1997) Accelerator driven systems:-energy generation and transmutation of nuclear waste, Status Report, November 1997.

IAEA-TECDOC-1083 (1999) Summary and future trends, status of liquid metal cooled fast reactor technology, April 1999.

Ion, S.E., Bull, A.J., Ivens, R. & Abram, T.T. (2003) Meeting Europe's energy needs through next generation nuclear systems – an industrial view of the role of EC co-sponsored research, FISA 2003, EU Research in Reactor Safety, Luxembourg 10–13, 2003.

IRSN: Scientific and Technical Report (2002) IRSN Communications Department, December 2002.

Jacq, F. & Allelein, H.-J. (2000) ASTEC V0.2 General Overview, ASTEC Report, Doc 99-09, March 2000.

Joo H.G. *et al.* (1998) Purdue advanced reactor core simulator, a multi-dimensional two-group reactor kinetics code based on non-linear analytical nodal method, PU/NE-98-26, September 1998.

Jovall, O., Larsson, J.-A., Shaw, P., Klinghoffer, O., Touret, J.-P. & Karlberg, G. Concrete containment management using the finite element technique combined with in-situ non-destructive testing of conformity with respect to design and construction quality (CONMOD), FISA 2003: EU Research in Reactor Safety, 2003.

Jumel, S., Domain, C., Ruste, J., Van Duysen, J.C., Becquart, C., Legris, A., Pareige, P., Barbu, A., Van Walle, E., Chaouadi, R., Hou, M. Odette, G.R., Stoller, R.E. & Wirth, B.D. (2000) Simulation of Irradiation Effects in Reactor Pressure Vessel Steels: the Reactor for Virtual Experiment – REVE Project, Effects of Radiation on Materials: 20th International Symposium, ASTM, STP 1405, Eds. Rosinski, S.T. & Kumar, A.S. American Society for Testing and Materials, West Conshohocken, PA, 2000, FISA 2003: EU Research in Reactor Safety.

Jumel, S., Domain, C., Monnet, G., Ruste, J., Van Ruysen, J.C., Bacon, D.J., Voskoboinikov, R., Devincre, B., Barbu, A., Christien, F., Perlado, M., Aravelo, C., Hou, M., Kulikov, D. & Tagstrom, P. Simulation of radiation effects in Zr–Nb alloys (SIRENA), FISA 2003: EU Research in Reactor Safety, 2003.

Lanning, D. *et al.* (1997) NUREG/CR-6534.

Lidbury, D.P.G., Bass, B.R., Gilles, P., Connors, D., Eisele, U., Keim, E., Keinanen, H., Lauerova, D., Marie, S., Nagel, G., Nilsson, K. & Wadier Y. Validation of constraint based methodology in structural integrity (VOCALIST), FISA 2003: EU Research in Reactor Safety, 2003.

MELCOR 1.8.4 Computer Code Manual (1998) vol. 2: Reference Manuals and Programmers' Guides, NUREG/CR-6119, vol. 2, Rev 1, SAND97-2398, May 1998.

Millard *et al.* (1992) Research and Development Performed at CEA/DMT in the Field of Nuclear Reactor Containments of the Fifth Workshop on Containment Integrity, NUREG/CP-0120, SAND92-0173, Washington, D.C., 12–14 May 1992.

Mitchell, C.H., Farrar, B., Lo Pinto, P., Dominguez, M.T., Van Heek, A., Every, D. & Carluec, B. (2001) Gas-cooled fast reactor concept review studies (GCFR), FISA 2001: EU Research in Reactor Safety, Luxembourg, 12–15 2001, EUR 20281.

NUREG/CR-5545, Rev. 1 (1992) VICTORIA: A mechanistic model of radionuclide behaviour in the reactor coolant system under severe accident conditions, December 1992.

NUREG/CR-6533 (1997) Code Manual for CONTAIN 2.0: A Computer Code for Nuclear Reactor Containment Analysis, December 1997.

NEA/CSNI/R(99)25 (2000) Fuel Safety Criteria Technical Review, July 2000.

NEA Annual Report (2002) 31st Annual report of the OECD Nuclear Energy Agency, France.

Page, R. *et al.* (1998) Description of the TALINK (Rev1.2) Code for Linkage of Codes Executing in Parallel, C5166/TR/050 Issue V02, February 1998.

Paillere, H., Garcia-Cascales, J., Kumbaro, A., Bestion, D., Mimouni, S., Stadtke, H., Franchello, G., Worth, B., Graf, U., Romstedt, P., Toro, E.F., Romenski, E., Deconinck, H., Ricchiuto, M., de Cachard, F. & Smith, B. Advanced three dimensional two-phase flow simulation tools for application to reactor safety (ASTAR), FISA 2003: EU Research in Reactor Safety, 2003.

RELAP5/MOD3 Code Manual (1995) NUREG/CR-5535, INEL-95/0174, June 1995.

Sammarato, R. *et al.* (1992) A general approach for containment success criteria under severe accident loads, *Proceedings of the Fifth Workshop on Containment Integrity, NUREG/CP-0120, SAND92-0173, Washington, D.C., 12–14 May 1992*.

Scheuerer, M., Heitsch, M., Menter, F., Egorov, Y., Toth, I., Latrobe, A., Pigny, S., Martin, A., Krepper, E., Willemsen, S., Muhlbauer, P., Andreani, M., Karlsson, nR., Henriksson, M., Karppinen, I. & Kimber, G. Evaluation of computational fluid dynamic methods for reactor safety analysis, (ECORA), FISA 2003: EU Research in Reactor Safety, 2003.

Smith, N.R., Shuttleworth, E., Grimstone, M.J., Hutton, J.L., Armishaw, M., Bird, A., France, N. (all Serco Assurance) & Connolly, S. (BNFL) The current status and future plans for the Monte Carlo Codes MONK and MCBEND, *Proceedings of Monte Carlo 2000*, Lisbon, Portugal, October 2000.

Spore *et al.* (2001) TRAC-M/FORTRAN 90 (Version 3.0) Theory Manual, NUREG/CR-6724, July 2001.

Van Duysen, J.C., Jumel, S., Boutard, J.L., Lidbury, D.P.G., Van Dyck, S. & Sevini, P. (2004a) Prediction of irradiation effects on nuclear reactor components (PERFECT), FISA 2003: EU Research in Reactor Safety.

Van Duysen, J.C., Domain, C., Jumel, S., Bacon, D.J., Devincre, B., Cleri, F., Perlado, M., Malerba, L. & Soisson, F. (2004b) Improvement of techniques for multiscale modelling of irradiated materials (ITEM), FISA 2003: EU Research in Reactor Safety.

Weiss, F.-P., Rohde, U., Karlsson, R., Hemstrom, B., Lillington, J.N., Scheuerer, M., Tuomisto, H., Toppila, T., Dury, T., Cvan, M., Muhlbauer, P., Toth, I., Elter, J. & Bezrukov, Y. Fluid mixing and flow distribution in the reactor circuit (FLOMIX-R), FISA 2003: EU Research in Reactor Safety, to be published.

Wichers, V.A., Malo, J.Y., Starflinger, J., Heitsch, M., Preusser, G., Tuomainen, M., Huggenberger, M. Testing and enhanced modelling of passive evolutionary systems technology for containment cooling (TEMPEST), FISA 2003: EU Research in Reactor Safety, 2003.

Wright, G.A., Shuttleworth, E., Cowan, P., Curl I.J. & Mattsson, C.G. (1999) MCBEND – A fluence modelling tool, *Proceedings of the International Symposium on Reactor Dosimetry ISRD'99*, Osaka, Japan, September 1999.

Yadigaroglu, G. Computational fluid dynamics for nuclear applications from CFD to multi-scale CMFD, FISA 2003: EU Research in Reactor Safety, 2003.

Chapter 17

The Future of Nuclear Energy

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Chapter 17
The Future of Nuclear Energy

17.1. INTRODUCTION/OBJECTIVES

This chapter will look at postulated future trends for energy requirements in the shorter and longer terms, extending over the next few decades. It will consider how nuclear energy could meet these requirements. It will also consider, albeit only briefly, some of the non-nuclear options that are being put forward as an alternative to nuclear power to meet demands. The chapter summarises the various applications of nuclear energy, including electricity generation, but also the other potential additional applications. It will provide a projection of possible nuclear development strategies in the industrialised and developing countries of the world. It will also bring together, in summary, the most important issues associated with the future of nuclear power that have been considered in the book.

Even without 'new build', many nuclear power plants will continue to operate for the next few decades and will offer a reliable carbon-free source of energy for electricity generation. However, fossil fuels are likely to occupy an increasing fraction of the energy supply, with the consequent issue of increased emissions of greenhouse gases. There will be newer technologies such as natural gas combined cycle plants and fluidised bed boilers becoming available. Greenhouse emissions may be reduced somewhat by newer plants and more efficient processes but carbon emissions will still be significant with these generators. Renewable energy offers a carbon-free alternative to nuclear energy, but the volume of supply would need to be substantially scaled up before it could replace the fraction of power generation currently produced by nuclear power.

17.2. FUTURE GLOBAL POWER REQUIREMENTS

The demand for energy is closely driven by economic growth. There are, therefore, significant differences across the global sectors. Data provided in Energy Visions 2030 for Finland (2003) show the emergence of countries such as Asia with large developing economies where the regional share of worldwide energy was 25% in 1981 but rising to 37% by 2005. Growth rates in Asia have been higher than other sectors since 1993, at about 4.8% between 1993 and 2000 and forecasted to exceed 4.5% between 2000 and 2005.

International Energy Agency (IEA) data forecast an average global annual growth rate of 3% over the next 20 years. This equates to about a 57% growth of primary energy requirement over this period. The main increase in demand will come from the developing

Table 17.1. Percentage of EU total energy consumption

Fuel	2000 (%)	2030 (%)
Oil	41	38
Gas	22	29
Coal	16	19
Nuclear	15	6
Renewables	6	8

Data from EC Green Paper (2000) and European Energy Strategy (2001).

countries. This demand is likely to be met from their indigenous resources of fossil fuels together with additional imported energy resource to meet demand. The fossil fuel share could be as high as 90% by 2020 unless this additional resource can be supplied by other means, e.g. nuclear, hydropower or possibly renewables.

Another forecast for the EU is little different (EC Green Paper, 2000; European Energy Strategy, 2001). The distribution of total energy consumption across the EU for the various sources is shown in Table 17.1. To meet this demand, Europe currently imports about 50% of its requirement, and this would rise to 70% in 2030 if current trends continue. Without new build of nuclear plants, the nuclear component would drop from 15 to about 6% in 2030, the European energy sector would become much less autonomous and without a significant increase in renewable energy, carbon dioxide emissions and global warming would increase.

17.3. ENERGY STRATEGIES

There have been relatively slowly changing trends in energy infrastructures over the past few decades but the mix of future energy providers is likely to change in the future. A number of countries are reviewing their energy policies for some time in the future. For example, the UK government published an Energy White Paper (Energy White Paper, 2003) in 2003, which proposed an energy policy looking forward to the year 2050. The paper covered all forms of energy requirement, from electricity generation, heating and lighting to transport, industry and communications. It was based on in-depth analysis following a report published by a UK-appointed strategy unit in 2002 (The Energy Review, 2002). Other countries are performing similar reviews, see e.g. the forward vision to 2030 published by VTT, Finland (Energy Visions 2030 for Finland, 2003). The strategy for the UK is outlined below, by way of example.

The major global challenges that need to be faced are:

- environmental and climatic change from carbon dioxide levels increase;
- decline of the world's indigenous energy supplies, from oil, gas, and coal and how these may be replaced (e.g. by nuclear, renewables);

- the need to update national energy infrastructures over the next few decades to meet new energy mixes.

The goals of most of the industrialised countries are to:

- reduce carbon dioxide emissions with specific targets. In the UK the goal is to cut carbon dioxide emissions by some 60% by about 2050, with significant progress by 2020. Many, but not all, countries support the Kyoto Protocol;
- maintain reliability of energy supplies;
- promote competitive markets, raising the rate of sustainable economic growth and improving productivity. There is an increasing trend toward deregulation;
- meet other energy (non-electrical) requirements for industrial and domestic supply (e.g. to ensure every home is adequately and affordably heated).

To meet these goals, it is likely that an energy system will be required that is quite different from that of today. Much more diverse systems are envisaged. These will include a balance between imported energy and fuel, a mix of large power stations, that could include offshore marine plants, including wave, tidal and wind farms and also onshore wind farms. There would be an increase in local generation, including biomass, local wind and tidal generators and micro-generation from combined heat and power (CHP) plant, fuel cells or photovoltaics. Energy efficiency improvements would be expected from improved home design. Gas might be expected to form a large part of the energy mix whereas coal fired generation would either play a reduced part or be linked to carbon dioxide capture and storage.

There have been debates in many sectors (industry, learned societies), etc. on how goals for security of energy supply can be achieved and there are many different opinions. In the UK for example, the future of energy was the focus of the 2002 Parliamentary Links Day, organised by the Royal Society of Chemists (http://www.rsc.org/lap/parliament/linksday.htm). This included an audience of distinguished scientists and politicians and covered energy-related activities taking place in government and industry. The scope was broad across the energy spectrum, covering nuclear and non-nuclear, electrical and non-electrical applications.

Many of the presently operating nuclear plants will be shut down over the next two decades. In the UK, by 2020, the existing AGR nuclear power stations will almost all have reached the end of their lives and all the Magnox stations will have shut down. However, new build continues in Asia and some new plants are likely in Europe in the next few years. Nuclear power remains an option for the future for the UK. However, the Government White Paper did not propose it and stated that before any decision to proceed with the building of a new power station, there would need to be the fullest consultation and publication of a White Paper setting down the Government's proposals. The arguments for a delay were both on economic grounds and concerned with the issue of waste disposal (sustainability).

It is increasingly recognised internationally (within the EC, US and Japan, as described in the section on hydrogen generation) that the ‘hydrogen economy’ has significant benefits as a clean and flexible energy system. In a report to the Parliamentary Links Day, the UK Government’s Chief Scientific Adviser also anticipates a significant move towards a hydrogen economy by 2020 (http://www.rsc.org/lap/parliament/linksday.htm). This view is also supported by the UK nuclear industry (Clegg, 2002) and others. The issue is how to produce hydrogen without releasing carbon dioxide.

In the Section 17.4, world events of recent years are examined. After that, a discussion is given on how these and other developments may shape developments in the near future. The remaining sections continue to look further into the future, covering likely developments over the next half-century.

17.4. NUCLEAR INDUSTRY AND THE RECENT PAST

In this section, recent past in taken to infer the last decade.

There has generally been an improvement of performance at many plants. This has been evident from a number of performance measures, e.g. from WANO indexes and in the US, the Institute of Nuclear power Operations (INPO) (Sinco, 2003). This has been driven by better leadership and improved plant management.

Another driver for improved performance has been the move towards deregulation of the electricity industries in some countries, e.g. the US and the UK. This has resulted in competition in the electricity markets between all providers, nuclear and non-nuclear.

The last decade has seen the shutdown of some nuclear plants, for both safety and for economic reasons. For example, first generation VVER plants operating in former Eastern Germany were shutdown, following re-unification, because of safety concerns. On the other hand, business decisions on whether to shutdown some plants prematurely have depended on the scale of cost liabilities being carried.

There have thankfully been no major accidents over the past decade but there have been several incidents that have not helped the cause of the industry. The finding of boric acid corrosion in the reactor vessel head in the Davis–Besse plant has resulted in increased inspection, longer outages, etc. Although not on a reactor, the Tokai-mura incident in a fuel handling plant in Japan has also caused some concern.

Despite some of these more negative aspects, there has been a decade of safe and reliable operation. Building of new power plant in Asia has continued. Particularly in the last five years there has been an increase in confidence in some countries in which the industry was beginning to stagnate and the possibility of new build is now under consideration. This is true for Finland and France in Europe and also in the US.

17.5. NUCLEAR INDUSTRY AND THE NEAR FUTURE

The nuclear industry may be unique among the industrialised industries in regard to the safety standards expected from it. Increasingly higher standards will be placed in the future. Having operated successfully (in the main) for over half a century and having met these standards, any lapse would be quickly seized upon. Thus the most important aim in the near future is to ensure that safe and reliable operation continues.

As observed earlier, there will be increasing emphasis on ensuring that the environment is protected from the operations of industries. The nuclear industry will have to meet increasingly stringent limits on radiological releases; it will need to pay greater attention to emergency preparedness planning, etc. As already stated, the measures that are being considered to reduce greenhouse gas release should benefit the case for nuclear power, which in this respect is a clean source of power.

For the de-regulated utilities, there is a need to create more investor confidence. In the US, for example there is evidence that investors now perceive the industry more positively. It is seen to be a stable industry in the more competitive market of today and can offer advantages over its competitors. In the recent years, low and stable operating costs have been realised. A number of large multi-unit sites are generating at a little under 2–2.2 cents per kW h (Sinco, 2003).

In many countries a commitment to new build will be a business decision against other generator alternatives. The target in the US is around \$1100 per kW if it is to be competitive with combined-cycle gas.

Another factor in engendering investor confidence is to ensure a stable predictable licensing process in order to reduce uncertainties for the owner/utility. There have been moves towards design certification in both Europe and the US, which is an important step. However, there are still areas of uncertainty, e.g. associated with the time taken to gain the plant operating licence, following construction.

17.6. NUCLEAR ENERGY APPLICATIONS

The role of nuclear energy for carbon-free power generation is recognised by a number of national and international bodies, e.g. as noted in the UK Energy Review White Paper (Energy White Paper, 2003). An EC green paper has also been published noting the contribution of nuclear energy in meeting Kyoto Protocol targets (NEA Annual Report, 2002). However, there remains doubt internationally whether nuclear energy is a sustainable energy source. This issue has recently been discussed at the World Summit on Sustainable Development (WSSD) in Johannesburg, South Africa and the Eighth Conference of UN Framework Convention on Climate Change (COP8) at New Delhi, India.

Looking forward, a much wider range of energy generation mix is anticipated compared with the present day. For some of the options, nuclear energy is a viable source of primary energy. Nuclear power could be used to electrolyse water and produce hydrogen, or indeed can and has already been used for a number of other heat applications.

In general there is increasing environmental awareness in all the major industrialised countries, not just on the issues associated with nuclear power. The population will become more aware of the challenge of climate change and the part they can play in reducing carbon emissions. The content of carbon in fuels will increasingly become a commercial differentiator if the cost of carbon is reflected in prices. This should promote more reliance on non-carbon producing energy generators.

17.6.1 Electricity Generation

Electricity generation is by far the most important civil nuclear energy application. This is likely to remain the case in the future, although some additional applications are envisaged, as discussed below.

There are marked differences across the major industrialised sectors in regard to future trends for nuclear power electricity generation. In Asia, modest expansion can be expected, in Europe, Finland is preparing for new build, but other European countries, e.g. Belgium and Germany are pursuing phase out policies. Nuclear power potential is being reconsidered in the US. Table 17.2 shows a relatively pessimistic scenario for nuclear power whereby no new power plants are built, beyond those already being built or firmly planned, together with the retirement of old plants.

Regarding nuclear power for either electrical or non-electrical generation, a key safety issue concerns the management of nuclear waste. Supporters of nuclear energy argue that the technical problems associated with waste disposal are solved, opponents do not agree. There are other commercial and practical issues such as: capital cost, market price of nuclear electricity and energy, and the risks, including liabilities and availability of an adequate skill base. All these will impact any decision for new build. It is worth noting that some experts assert that the capital cost of modern nuclear plant is no higher than that of new coal plant. There are also predictions that the total cost of nuclear electricity of Generation IV reactors will be less than that of gas plant.

Table 17.2. Percentage change of nuclear power generation compared with 2001

Country Group	2010 (%)	2015 (%)	2020 (%)
North America	+2	−3	−6
Western Europe	−7	−13	−31
Eastern Europe	+12	+22	+23
Far East	+39	56	54
World total	+8	+9	+2

Data from Nuclear Technology Review (2003); +, increase, −, decrease.

In order to improve on energy efficiency, there is likely to be increased interest in CHP. For example, in the UK, about 9 GW of nuclear plant will be decommissioned over the next two decades, and by 2010 the UK is planning to install about 10 GWe of CHP plant (http://www-tec.open.ac.uk/eeru/naatta/renewonline/rol39/11.htm). This is a commitment in the Energy White Paper (Energy White Paper, 2003). Currently heat produced in electricity generation is largely wasted. CHP plants could be made to produce heat as well as electricity in approximately equal proportions. Supporters of non-nuclear energy generation argue that the adoption of gas-fired CHP plants would release gas currently used for heating, for use in electricity generation without leading to increase in carbon emissions. However, if nuclear plant provides the CHP energy source, then carbon emissions are quantitatively reduced.

17.6.2 Heat Applications

Nuclear heat is already being used for various direct heating applications (IAEA-TEC-DOC-1056, 1998). Although the primary utilisation of nuclear power has been and is likely to be for electricity generation, interest in heat applications is growing. Co-generation of heat and electricity and dedicated heating reactors have already been established, particularly in Russia. Operational experience exists on over 60 reactors supplying heat for district heating, seawater desalination and other industrial processes. The utilisation to date has been generally for low-temperature applications.

Reactor designs are being further developed for co-generation, district heating, seawater desalination and low-temperature process heat. These include water-cooled, PWR systems but also more innovative technologies including lead–bismuth reactors.

High- and medium-temperature applications are less well-advanced and have only been developed at laboratory or small scale. There are extensive programmes for high-temperature helium gas reactors which could be used for process heat applications that require high temperatures, e.g. processes that include oil refinement, coal gasification and also hydrogen generation (covered more elaborately in Section 17.6.3). High-temperature reactors (HTRs) could also be used in co-generation mode for district heating and desalination.

The IAEA is supporting various activities in promoting advanced nuclear energy heat applications (Nuclear Power, IAEA). The International Working Group on Gas Cooled reactors met in the UK in September 2002 to review activities in the field and make recommendations for future efforts. The Group noted that gas turbine high-temperature reactors currently under development are well suited to desalination, operating in a co-generation mode.

17.6.3 Hydrogen Generation

For small-scale power generation, it is anticipated that hydrogen fuel cells will be playing a greater part in the economy, initially in a static form in industry or as a means of storing energy. The hydrogen would be generated by non-carbon electricity. Hydrogen can be produced in many ways, e.g. renewable energy sources such as hydro, solar, wind power, electrolysis, biomass and by nuclear energy. Nuclear power could be used to provide electricity for electrolytic hydrogen production. Fuel cells could also be used to back up intermittent renewables. Fuel cells are an area of active research (N.B. in addition to hydrogen, it should be noted that biofuels are another possible option for fuel cells).

Transport is still a major contributor to air pollution and carbon dioxide emissions (about 30%). For transport, hydrogen could be increasingly used for fuelling public service vehicle fleets and utility vehicles and is, therefore required as a primary source. It could possibly be used in the car market where hybrid internal/combustion/electric vehicles would be commonplace in the car and light goods sectors. N.B. For these there is also likely to be a substantial and increasing use of low carbon biofuels. (It is worth noting that other innovative technologies are being investigated for transport, e.g. vehicles powered with batteries that can be charged by electromagnetic induction from metal plates buried in the road at selected stops.)

There is an increasing interest in hydrogen as an energy system, produced from a carbon dioxide free process (The Parliamentary Office of Science and Technology, &, Millbank, London, 2002). Hydrogen may have a number of widespread applications as a fuel for road transport, distributed heat and power generation and for energy storage. The most likely use for hydrogen in the UK and in other countries, is for transport, for fuelling fleet vehicles and buses. The Energy Saving Trust (EST) (The Parliamentary Office of Science and Technology, &, Millbank, London, 2002) refers to the use of hydrogen in fuel cell vehicles as 'the most promising option for zero carbon road transport'. The Institute for Public Policy Research (IPPR), an UK think-tank and the Carbon Trust, a non-profit company set up by Government to take a lead on low carbon innovation in the UK, are supporting the case for a high-level strategic approach towards developing a hydrogen economy (The Parliamentary Office of Science and Technology, &, Millbank, London, 2002).

There are a number of international initiatives towards developing the hydrogen economy including IEA, EC and OECD activities (http://www.iea.org/workshop/2003/hydrogen). There are major international activities in train, the EC has announced a large programme on hydrogen and renewable technologies, the US is supporting a five-year programme on hydrogen, fuel cells and related infrastructures and the Japanese have substantially increased their level of activity on hydrogen research since 1995 (http://www.iea.org/workshop/2003/hydrogen).

The EC has set up a high-level Group to assess the prospects for using hydrogen and fuel cells in transport and overall energy policy (http://www.world-nuclear.org/news/2002/wd_oct18.htm). The EU Clean Urban Transport for Europe programme aims to provide fuel cell buses in 10 European cities in the near future, including 3 in London (The Parliamentary Office of Science and Technology, &, Millbank, London, 2002). Also there is a European Integrated Hydrogen Project (EIHP) which aims to create a harmonisation of necessary legislation in the EU for hydrogen safety, infrastructure and standardisation (http://www.world-nuclear.org/news/2002/wd_oct18.htm).

In the UK, the Engineering and Physical Sciences Research Council (EPSRC) which funds a UK hydrogen energy network, also promotes hydrogen research (The Parliamentary Office of Science and Technology, &, Millbank, London, 2002). There are calls for a dedicated programme to co-ordinate and support UK research initiatives and support demonstration projects. The use of hydrogen as a fuel for buses is being pursued in the Cambridge Urban Solar Hydrogen Economy Realisation Project (The Parliamentary Office of Science and Technology, &, Millbank, London, 2002). Hydrogen fuel cells are being developed for local heating and energy supply applications.

17.6.4 Partitioning and Transmutation

The proliferation of plutonium and the threat from terrorism in modern society is a major driver towards a closed fuel cycle. Another driver is to develop a process for effective management of spent fuel and waste. Advanced reactor concepts provide a solution to these requirements.

For many years fast reactors have offered the attraction of a sustainable fuel supply based on a uranium–plutonium fuel cycle. Uranium resources will last for at least 60 years; so from this perspective there is no immediate need for fast breeder reactors, which (in addition) are about 50 times more efficient than current thermal reactors. There is now a current interest in exploring particular advantages of the fast reactor to consume plutonium, and reduce the stockpile of weapons fuel. Also the fast reactor can be used to irradiate minor actinides (MA) and fission products to reduce the toxicity of long-term wastes.

There are a number of international programmes at the present time that aim to develop the above technology. There are EC initiatives in this area; e.g. a review of gas cooled fast reactor concepts (Mitchell *et al*., 2001) was carried out within the Fifth Framework programme. The review partners concluded that the gas-cooled fast reactor (GCFR) has a number of potential advantages to offer.

The EC CAPRA (Consummation Accrue de Plutonium dans les reacteurs Rapides) project originally focused on technologies to consume existing plutonium stocks arising from the operation of commercial reactors (IAEA-TECDOC-1083, 1999). Work is currently underway in the EC CAPRA/CADRA project to evaluate the potential for the transmutation of plutonium and MA from waste. A wide variety of reactor concepts of metal cooled fast reactors (Smith *et al*., 2003; Hesketh, 2003; Vasile *et al*., 2001) are being considered. The aim is to transmute these actinide species to species with much shorter half-lives.

There are also various international activities on the application of proton particle accelerators in connection with subcritical reactor systems as a means of separating and eliminating actinides via transmutation.

Reactor systems for plutonium burning and the partitioning and transmutation of nuclear waste are among those selected for development within the Generation IV initiative.

17.6.5 Space Applications

Space reactor systems have been studied since the early days of nuclear power in the late 1950s. However, only one US reactor (SNAP-10A) (Harman and Susnir, 1964) and a few Russian reactors have ever been in space. There is now some renewed interest in nuclear power for space missions, in the US and also Europe.

In general, for space applications, fast reactor gas or liquid metal cooled designs, operating at high temperature are the most appropriate to meet the various requirements and in particular, launch constraints. Clearly also reliability is important and this depends on the status of the possible technologies.

Space applications include, planetary base applications, e.g. for Mars or the moon, nuclear propulsion and radioisotope power systems (RPS). For the former, possible designs include the lithium liquid metal cooled concepts, SP-100 in the US (Sapir *et al*., 1987) and the ERATO system in France (Carré *et al*., 1987), these generating power in the range 100–500 kWe. Gas cooled systems include the Sandia National Laboratories Dual Purpose design (Lipinski *et al*., 1999), and a Russian Project 1172 gas-cooled design (Andreev *et al*., 2000). A low-power PWR water-cooled system has also been investigated by Technicatome.

For propulsion, many of the reactor concepts under consideration have been developed from other applications. In general many reactor systems that have been developed to supply electrical power, can be employed as a power source in a nuclear electric propulsion (NEP) systems. The SP 100 and ERATO system could be adapted. The UK 200-SNPS was a particle bed system, designed for earth orbit electrical power supply, but could be adapted. The Enabler NERVA (Livingston and Pierce, 1991) was primarily aimed at nuclear thermal propulsion (NTP), where the energy source heats the propellant directly (as opposed to NEP where electrical power from the reactor is used for accelerating the propellant). The Russian TOPAZ-2 liquid metal (NaK) cooled system (Voss *et al*., 1991) or more advanced TOPAZ concepts could be used. There are also combined cycle (NEP&NTP) nuclear propulsion and other advanced concepts under consideration.

Some of the reactor designs are such that the same generic design can be used for both planetary base and propulsion applications. An example of one such is the ESCORT Derivative reactor (Feller and Joyner, 1999), designed for in-space propulsion and power (25 kWe) and to supply 160 kWe for 10 years on the surface of Mars.

Finally RPS consisting of a nuclear radioisotope heat source and power conversion, have been developed. This technology started in the SNAP programme in the 1950s and culminated in the General Purpose Heat Source (Angelo and Buden, 1985) module flown on the Galileo and Ulysses spacecraft. RPSs typically generate a few kilowatts.

Space nuclear reactor programmes are being supported by the National Aeronautics and Space Administration (NASA) (Nuclear Reactors in Space) and the European Space Agency (ESA) programme. A review of space nuclear power and propulsion for future space exploration is given in (Bond and Sweet, 2003). A particular interest at present is the benefits of nuclear power systems for Mars exploration (Sweet *et al*., 2002). In particular, work is on-going to examine the feasibility of different reactor systems, including the feasibility of a small gas-cooled, particle bed reactor, to power a Mars mission.

17.6.6 Other Small Reactor Applications

Miscellaneous small reactors are needed for many different applications including materials testing and irradiation, isotope production, and reactor and nuclear physics training. Further applications include neutron detector calibration, neutron activation trace element analysis and delayed neutron counting for evaluating fissile content and basic research applications.

A matter of growing concern is the reducing numbers of such reactors that remain in service. However, many of these reactors are ageing and are approaching 50 years of life. They are, therefore, reaching the end of their operational lives. In particular, the EC is currently evaluating the future needs of material test reactors in Europe (Parrat *et al*., 2003) which provide valuable services within Europe and worldwide. Materials testing facilities are likely to be needed for the development of some of the advanced Generation IV concepts that will include corrosive materials resulting in chemically and physically demanding environments.

Most of the therapeutic isotopes required by industry are currently produced using neutron irradiation in research and small reactors. However, with a potential 10-fold intensity increase in compact cyclotrons, some charged particle reactions are becoming accessible for producing some of the newer isotopes. Reliance on research reactors may diminish as accelerator-based techniques are developed and able to provide adequate technical capability at prices industry can support (Lewis).

There are many novel applications of nuclear energy in medicine at various stages of development. Examples include boron neutron capture therapy (BNCT), a technique being pioneered at Birmingham University for the treatment of cancer. This involves injecting boron into the patient, which concentrates in the affected organ and which is then irradiated. Another example involved a technique that has recently been applied in Italy, where a patient with liver cancer, had the organ removed, irradiated and replaced with successful remission of the tumour.

There are fewer nuclear engineering degree courses now available at the Universities and fewer small reactors available for teaching purposes. In the UK, collaborative research programmes between academia and industry are being undertaken by the University of Birmingham. Current projects in the Nuclear Physics Group relate to modelling of nuclear materials assay equipment and the study on nuclear waste transmutation (http://www.np.ph.bham.ac.uk/research/npt.htm). Academic research is conducted at the Imperial College research reactor, situated in Silwood Park (http://www.imperial.ac.uk/publication/pbb/env_sci/intro.htm).

Other interesting applications of nuclear energy concern topics such as food irradiation. This is a growing international business (www.sercoassurance.com/answers). The process involves the use of high-energy gamma radiation, produced by a source, to kill bacteria in food and preserve it. Other possible applications include sterilisation of materials and implements for the medical industry.

Small reactors operate with different fuel cycles compared with large power reactors. There are research reactors of diverse design in a number of countries, including Australia (heavy water), India (pool type) and Japan (fast reactor) (http://www.world-nuclear.org/info/inf61.htm).

17.7. ADVANCED NUCLEAR REACTOR TECHNOLOGIES

17.7.1 Water Reactors

Light water reactors are the most widely used type of reactor in service at the present time and much work is taking place in optimising the performance and safety of advanced evolutionary designs. A similar approach is being adopted in the development of evolutionary heavy water reactors The emphasis has been to improve the operating economics and also to simplify design to reduce construction costs.

Recent focus has been on large power generation (1300–1500 MWe) but smaller and medium-sized plants are in consideration. There is an increased tendency to introduce more passive systems but some passive safety systems are less appropriate for large power generation.

More innovative types of water reactors are being considered within the first phase of the Generation IV programme (Gen IV-A) (Sinco, 2003). The supercritical water-cooled reactor (SCWR) is a high-temperature super-critical pressure reactor that could be developed from present water reactor technology (Overview of Generation IV Roadmap). It would be primarily for electricity generation. However, there are two core design options, offering an open fuel cycle with a thermal spectrum or a closed fuel cycle with a fast spectrum to enable actinide management. The projected time for commercial deployment of the thermal spectrum option is around 2020–2030. Table 17.3 shows approximate timescales for the different advanced nuclear reactor technologies.

Table 17.3. USDOE projection of power plant developments

Time period	Events
2005–2010	Optimisation of nuclear plant Continue to operate existing plant
2010–2020	Deploy first US ALWR
2020–2030	Deploy first-phase commercial Gen IV-A thermal reactor
2030–2050	Deploy second phase commercial Gen IV-B fast reactors
2050 +	Fusion

Data from Sinco (2003).

17.7.2 High-Temperature Gas Reactors

Gas reactors have been operating successfully in the UK over many years. High-temperature gas reactors has been operated in the UK, US and Germany and new smaller plants are in operation in China and Japan. There is a revived interest in HTRs and in particular the South African Pebble Bed Modular Reactor (PBMR) (Clegg, 2002; http://www.bnfl.com/website.nsf/researchmenu.htm; Hittner, 2002). R&D activities can, therefore, build on considerable previous experience.

The very-high-temperature reactor (VHTR) system is another one of the thermal reactor types under consideration in the Gen IV-A programme (Sinco, 2003). It would operate at very high core outlet temperatures, >1000°C and have very high efficiency compared with current generation plant. It could be used for electricity generation or hydrogen production using water cracking technology (Overview of Generation IV Roadmap). It could also be used in the process heat and chemical industries. A timescale of 2020–2030 is envisaged for commercial deployment of these systems.

17.7.3 Gas-Cooled Fast Reactor

GCFR systems have been considered in the past but were not developed. The GCFR is now being considered as a longer-term option in the second phase Generation IV programme (GEN IV-B) (Sinco, 2003).

The GCFR is one type of fast neutron system that is being put forward for use in a closed fuel cycle, thereby reducing the problem of long-term proliferation concerns (Overview of Generation IV Roadmap). The GCFR shares many of the attributes of the high-temperature thermal reactor with high outlet temperatures, enabling efficient electricity generation, hydrogen production or process heat applications. It has the additional benefit of enabling the full recycle of actinides minimising long-lived radioactive waste. Being a fast spectrum, it would utilise fissile and fertile fuel more efficiently than the high-temperature thermal systems with a once-through fuel cycle. Fast spectrum systems such as those based on GCFR technology are not expected to be available as early as the thermal systems. GCFRs are projected to be available commercially towards 2030–2050.

17.7.4 Sodium-Cooled Fast Reactor

Sodium-cooled fast reactor (SFR) technology has been established over several decades and medium-scale prototype plants have been built and operated in several countries, including, e.g., France, UK, and elsewhere.

The SFR is also a Gen IV-B technology which is being put forward at both a medium- and large-size scale (Sinco, 2003). It is seen at present as mainly for the management of plutonium and other actinides and high-level waste (Overview of Generation IV Roadmap). As with all fast spectra systems, it offers an efficient utilisation of fissile and fertile materials in a closed fuel cycle. It is possible that it could be used as an electricity generator but at present capital costs are too high. A 2030–2050 timescale is again the projected timescale for the commercial SFR.

17.7.5 Lead-Cooled Fast Reactor

The lead-cooled fast reactor (LCFR) is another Gen IV-B system (Sinco, 2003). It utilises lead or lead–bismuth eutectic cooling in a fast spectrum system with the attributes of full actinide recycle fuel cycle and efficient conversion of fertile uranium (Overview of Generation IV Roadmap). It offers the prospect of a very long core life up to around 30 years with the obvious proliferation benefits.

It could be put forward at a range of different ratings from a small ‘battery’ scale, a medium-scale modular version, or a large scale of the greater than 1000 MWe range. It, therefore, offers a flexible option for distributed generation of electricity on small grids and for other energy products, including hydrogen products or desalination, through to large-scale electricity generation. The LCFR requires significant materials advancements for application in corrosive high-temperature environments. It is not expected to be available commercially until around the 2030–2050 timescale.

17.7.6 Molten Salt Reactor

The molten salt reactor (MSR) is another Generation IV technology. It offers a full actinide recycle within an epithermal spectrum reactor system (Overview of Generation IV Roadmap). It is envisaged as a large-scale plant of the order of 1000 MWe operating with a high outlet temperature with therefore good thermal efficiency. It is a flexible system offering efficient utilisation of plutonium and MA management. As currently envisaged, there is a relatively complicated heat exchanger system with a large number of sub-systems. Therefore, the economics are less favourable than for some of the other future plants that are being proposed. Its main application would be for electricity generation and plutonium and actinide destruction. The timescale for a commercial plant would also be around 2030–2050.

17.7.7 Accelerator Driven Systems

Accelerator driven systems (ADS) are hybrid systems combining a subcritical reactor together with a high-energy particle accelerator in order to produce a self-sustained reaction. ADS can be designed for both fast and thermal neutrons systems. They can utilise different fuel forms (solid, liquid), different fuel cycles, and different coolants and moderators. These have similarities with corresponding critical reactor systems, both in terms of the materials used and the applications that are possible. The objective of some ADS is the nuclear transmutation of Pu and MA in waste, with or without energy production; the objective in others is to utilise the thorium fuel cycle for energy production (IAEA-TECDOC-985, 1997).

Fast neutron systems are available with U/Pu solid fuel cycles, Na or Pb cooled, also with U/Pu liquid fuel with molten chlorides or Pb/Bi; both being suitable for MA incineration. The Th/U solid fuel cycle is Pb cooled and suitable for energy production or waste transmutation. Thermal ADS include solid Pu fuel systems with heavy water, for Pu weapons burning. There are quasi-liquid U/Pu graphite particle beads systems, He/heavy water-cooled, for MA management. There are liquid fuel systems encompassing U/Pu with molten salt for Pu, MA and FP management; Th/U with molten salt for energy production and U/Pu with heavy water for MA and FP transmutation, and energy production. Most concepts are based on linear accelerators, but some on a proton cyclotron concept.

ADS have some advantages and some disadvantages compared with critical reactor systems (NEA/OECD Expert Group Study, 2002). In terms of advantages, they allow the possibility of operating with a neutron multiplication factor of less than unity. They can be designed as pure transuranics (TRU) or MA burners and therefore would minimise the fraction of dedicated transmutors on a site. Reactor power is proportional to accelerator current, which simplifies control. From a safety perspective, the reactivity margin to prompt criticality can be increased, without dependence on delayed neutrons. Excess reactivity can be eliminated, allowing more flexibility in core safety design.

With regard to disadvantages, there is a reduction in net plant efficiency and the overall plant is more complex. The accelerator must have high reliability against thermal shocks. There are extreme stress, corrosion and irradiation loads on the beam window and target. There is also increased power peaking because the neutron source is external. There are compromises that have to be made between the neutron multiplication factor and the power produced. From a safety point of view, there are new types of reactivity and source transients that need to be taken into account, because the external neutron source can vary rapidly and the feedbacks from TRU and MA cores are weak.

Finally, in this last chapter, a few comments are made on the status of fusion research.

17.7.8 Fusion

The fusion reactor is still on the horizon for long-term energy generation. It is difficult to forecast the timescales for the development of the technology as a commercial power source. The UK Energy White Paper anticipates that nuclear fusion will be at an advanced stage of research and development by 2020 (Energy White Paper, 2003). Other commentators believe the reactor will still be in the development phase by 2030 (Energy Visions 2030 for Finland, 2003). Commercial realisation is unlikely to be before 2050 + . The fusion reactor is more attractive as a sustainable energy resource than the fission reactor since there are limitless fuel resources, there are no long-lived nuclides in the waste produced and the worst accident situations are of relatively low consequence.

Fusion research has and is being conducted in a number of collaborative international programmes. During the 1990s, the Joint European Torus (JET) project has made progress in generating significant amounts of energy. For the future generation of Tokamaks, interested nations will participate in the International Tokamak Experimental Reactor (ITER) project.

17.8. SUMMARY

Even without the building of new nuclear plants, IAEA projections indicate that global nuclear generation will continue at least at the present level or higher, until around 2020. Large decreases in Western Europe and to a lesser extent in the US will be compensated by significant increases in the Far East and to a lesser extent in Eastern Europe.

Decisions on nuclear power continuation will be country dependent and will depend on the perceived benefits against the risks and alternatives for other forms of energy generation. There will be strong economic competition from the fossil fuel generators, e.g. combined cycle gas plants.

In deregulated industries, for nuclear new build, there will need to be frameworks in place to enable power companies to accept their large capital investment risk, in particular for them to have confidence that building cost forecasts and construction schedules can be met. There is also the issue of long-term operational risk (stability of electricity prices) and eventual decommissioning costs. Finally, risks arising from delays in the regulatory licensing process must be acceptable. There is progress in some countries towards resolving these issues, e.g. in the US.

For new build, there will most likely be a need for a nuclear obligation from governments to enable suppliers or operators to sign up for long-term contracts. It will also be necessary to put in place some kind of Price–Anderson act to limit insurance risks.

Another important factor regarding the continuation of nuclear power will be whether an acceptable solution to the legacy and future waste problem becomes available. Further, utilities will probably require some type of fixed price contract from governments for managing their waste, i.e. governments will have to accept liabilities for waste.

The long-term future of nuclear energy may be influenced by increased global environmental legislation to limit carbon emissions, if the rate of ‘greenhouse’ gases continues to rise. On the assumption that the latter does occur, nuclear power will need to compete for acceptance against alternative carbon-free (renewables) energy generators. The economic case will depend heavily on whether there exist carbon premiums on generation, e.g. carbon taxes or permits.

REFERENCES

Andreev, P.V. *et al*. (2000) Atomic power stations for Mars, vol. 11 ISTC Project No. 1172 Report: Preliminary Project of the Manned Mars Expedition, September 2000.
Angelo, J.A. & Buden, D. (1985) *Space Nuclear Power*, Orbit Book Company, Malabar, FL.
Bond, R. & Sweet, D. (2003) Review of space nuclear power and propulsion for future space exploration, SA/PSS/124505/W001, January 2003.
Carré, F. *et al*. (1987) Status of CEA design and simulation of a 200 kWe turboelectric space nuclear power system, Fourth Symposium on Space Nuclear Power Systems, Albuquerque, NM.
Clegg, R. (2002) BNFL, The gap in UK Fission R, D & D, Presentation to the Energy Choices Conference, London, 5th December, 2002.
EC Green Paper (2000) Towards a European Strategy for the Security of Energy Supply, COM (2000) 769.
Energy Visions 2030 for Finland, (2003) *VTT Energy*, Edita Prima Ltd, Helsinki.
Energy White Paper, The Licensing Division, HMSO, St Clements House, 2–16, Collegate, Norwich, NR3 1BQ, Crown Copyright 2003.
European Energy Strategy, (2001) *Nucl. Energy*, **40** (2), 66–78.
Feller, G.J. & Joyner, R. (1999) Pratt and Whitney ESCORT Derivative for Mars Surface Power, CP458 Space Technology and Applications International Forum, NM.
Harman, T. & Susnir, J. (1964) SNAP-10A Reactor Design Summary, Rev. 1, NAA-SR-8679 (Rev).
Hesketh, K. (2003) Advanced fuel designs for existing and future generations of reactors: driving factors from technical and economic points of view, FISA 2003: EU Research in Reactor Safety, Luxembourg, 10–13 November, 2003.
Hittner, D. (2002) HTR-TN and the European programme for the development of HTR technology, ENS Conference, Lille, 2002.
http://www.bnfl.com/website.nsf/researchmenu.htm.
http://www.iea.org/workshop/2003/hydrogen.
http://www.imperial.ac.uk/publication/pbb/env_sci/intro.htm.
http://www.np.ph.bham.ac.uk/research/npt.htm.
http://www.rsc.org/lap/parliament/linksday.htm.
http://www-tec.open.ac.uk/eeru/naatta/renewonline/rol39/11.htm.
http://www.world-nuclear.org/info/inf61.htm.
http://www.world-nuclear.org/news/2002/wd_oct18.htm.
IAEA-TECDOC-985 (1997) Accelerator driven systems – energy generation and transmutation of nuclear waste, Status Report, November 1997.
IAEA-TECDOC-1056, (1998) Nuclear heat applications: design aspects and operating experience, November 1998.
IAEA-TECDOC-1083, (1999) Summary and future trends, status of liquid metal cooled fast reactor technology, April 1999.

Lewis, D.M. Accelerators in Industry, Amersham Laboratories, Bucks, HP7 9LL.
Lipinski, R.J., Wright, S.A., Lenard, R.X. & Harms, G.A. (1999) A gas cooled reactor surface power system, CP458 Space Technology and Applications International Forum, NM.
Livingston, J.M. & Pierce, W.L. (1991) The enabler – based on proven NERVA technology, Eighth Symposium on Space Nuclear Power Systems, Albuquerque, NM.
Mitchell, C.H., Farrar, B., Lo Pinto, P., Dominguez, M.T., van Heek, A., Every, D. & Carluec, B. (2001) Gas-cooled fast reactor concept review studies (GCFR), FISA 2001: EU Research in Reactor Safety, Luxembourg, 12–15, 2001, EUR 20281.
NEA Annual Report (2002) Trends in Nuclear Power, http://www.nea.fr/html/pub/activities/ar2002/nea-ar-2002-trends.pdf.
NEA/OECD Expert Group Study (2002) Accelerator driven systems (ADS) and fast reactors (FR) in advanced nuclear fuel cycles, a comparative study.
Nuclear Power, IAEA.
Nuclear Reactors in Space, http://www.uic.com.au/nip82.htm.
Nuclear Technology Review, 2003 Update, IAEA.
Overview of Generation IV Roadmap, www.nei.org/documents/DOE_Gen_IVDiagrams.pdf.
Parrat, D., Turnbull, J.A., Harbottle, J., Vitanza, C., Gubel, P., Kysela, J., Becquet, M., Manzel, R. & Pascal, C. (2003) Future needs for material test reactors in Europe (FEUNMARR), FISA 2003: EU Research in Reactor Safety, Luxembourg, 10–13, 2003.
Sapir, J. *et al.* (1987) Conceptual design of a reactor for a 300 kWe space nuclear power system, Fourth Symposium on Space Nuclear Power Systems, Albuquerque, NM.
Sinco, P. (2003) A look back, a look ahead, ANS Utility Meeting, *Nuclear News*, October 2003.
Smith, P.J., Hosking, J.G. & Newton, T.D. (2003) Fuel heterogeneity in a particle fuelled ETGCFR core design, SA/PSS/14339/W003 Issue 1, April 2003.
Sweet, D.W., Bond, R.A., Smith, P.J., Wright, G.A. & Bond, A. (2002) Nuclear power systems for MARS exploration, ENS Conference, Lille, France 2002.
The Energy Review, A Performance and Innovation Unit Report, Cabinet Office, Admiralty Arch, London, SW1A 2WH, February 2002.
The Parliamentary Office of Science and Technology, &, Millbank, London SW1P 3JA, 2002, www.parliament.uk/post/home.htm.
Vasile, A. *et al.* (2001) Fast reactor fuel cycle: core physics results from the CAPRA-CADRA programme, *Global 2001 Proceedings*, *Paris 9–13 September 2001*.
Voss, S.S. *et al.* (1991) TOPAZ-2, thermionic space nuclear systems and perspectives of its development, Eighth Symposium on Space Nuclear Power Systems, Albuquerque, NM.
www.sercoassurance.com/answers.

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