Haydee Domenech

# Radiation Safety

## Management and Programs

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*To my kids I have always loved so much…*

# Preface

This book is focused on the practical aspects of radiation safety organization, and is intended for a broad range of professionals in industry, research, education, agriculture, and medicine.

After an introduction to the interaction of ionizing radiation with matter, this book calls attention to important physics fundamentals of the discipline with definitions of quantities and units, current approaches to dose and shielding calculations, details of practical use of existing measurement instruments, and refers to the concerns about the effects of low doses of ionizing radiation. Furthermore, this book discusses the requirements for the different exposure situations according to the most updated recommendations of the International Commission of Radiological Protection (ICPR 103) and the International Basic Safety Standards for Radiation Protection (IAEA GSR Part 3 2014), and emphasizes the role of optimization and a well-structured and controlled Radiation Protection Program in building a strong safety culture, and seeking levels of dose As Low As Reasonably Achievable (ALARA).

This book also highlights the relevant legislation and regulations governing Radiation Safety in USA, and incorporates other topics of interest, including: radioactive waste management practices and transport of radioactive materials regulations as part of the control of public exposures, the importance of the assessment of potential exposures of workers and the public, data on accidents and their consequences, and a general overview of emergency planning and preparedness.

Finally, the book focuses on the practical aspects of radiation safety organization with examples of radiation protection programs for common applications such as: product irradiation, industrial radiography, well logging, nuclear gauges, custom and border inspection, and laboratories with unsealed radioactive sources.

Miami, USA — Haydee Domenech

# Acknowledgments

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# Contents

# Chapter 1
# What Does Ionizing Radiation Mean?

What is ionizing radiation? Where does it come from? What is it made of? To start answering these questions it is important to remember some atomic physics basics and how ionizing radiation interacts with matter.

Since 1913 when Danish physicist Niels Bohr proposed his atom model, it was established that all matter is made up of very small and invisible units named atoms, which are in turn composed of a dense center identified as the nucleus and a number of smaller negative electrically charged particles known as electrons, orbiting the nucleus in specific shells restricted to certain discrete values of energy. As the number of electrons increases, the atomic radius increases. This model was the beginning of quantum mechanics, which successfully explained many of the properties of atoms. The oxygen atom shown in Fig. 1.1 is an example of this model.

The nucleus, within which the bulk of the atom mass is concentrated, is now understood to be a quantum system composed of protons and neutrons, particles of nearly equal mass and the same intrinsic angular momentum (spin) of ½, both made of quarks and held together by the strong force of gluons. Protons and neutrons are known as nucleons. The proton distribution of the nucleus can be characterized by an average radius of $10^{-15}$ m. This radius is much smaller than that of the atom, which is typically $10^{-10}$ m. Thus, the nucleus occupies an extremely small volume inside the atom. Protons are electrically charged in a magnitude equal to the electrons, yet opposite in sign. The proton mass is far beyond the electron mass.[1] Neutrons do not carry any electrical charge.

Atoms in nature are generally electrically neutral, meaning that negative charges of the orbiting electrons are compensated by the same amount of positive charges of protons in the nucleus. However, to recognize an atom we have to take into account the neutrons in the nucleus. Any nucleus $X$ is then identified by its mass number

[1]Electron mass is $0.000910938188 \times 10^{-27}$ kg. Proton mass is $1.67262158 \times 10^{-27}$ kg.



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(denoted by $A$), which is the sum of protons and neutrons, and by its atomic number (denoted by $Z$), which is the number of protons. A nucleus of a certain atomic number and specific mass number is known as nuclide. The convention for designating a nuclide is by its atomic number $Z$ and mass number $A$ ($^{A}_{Z}X$). The oxygen atom illustrated in Fig. 1.1 has 8 protons and 8 neutrons: $Z = 8$; $A = 8 + 8$.

Nuclides from the same element having the same atomic number and different mass number are called isotopes. For example, there are three different isotopes of the element hydrogen with mass number 1, 2, and 3, respectively, and all of them have the same atomic number: $Z = 1$.

All nuclei in nature are either stable or unstable. Nuclear stability depends on the proton–neutron ratio and the mass number. Most light stable nuclei (up to about atomic number 20) have a neutron-to-proton ratio equal to 1. As the number of protons increases, the ratio of neutrons to protons necessary to ensure a stable nucleus increases steadily to about 1.5. Heavier nuclei—with mass number greater than about 100—have an excess of energy which made them unstable. Many isotopes are unstable, especially beyond atomic number 83.

Unstable nuclei tend to reach stability by emitting particles and electromagnetic radiation. Heavier nuclei tend toward stability by ejecting part of its mass converted into energy. All unstable nuclei are known as radioactive nuclides, radioactive isotopes or radionuclides. Of the three isotopes of hydrogen mentioned above only one, tritium ($^{3}$H) is radioactive. Tritium is produced as a result of cosmic reactions and, artificially, in nuclear reactors.

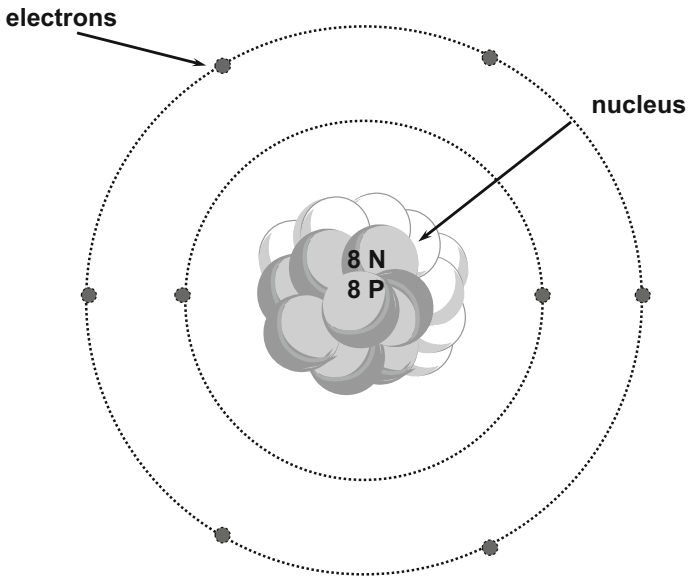


**Fig. 1.1** Model of the oxygen atom

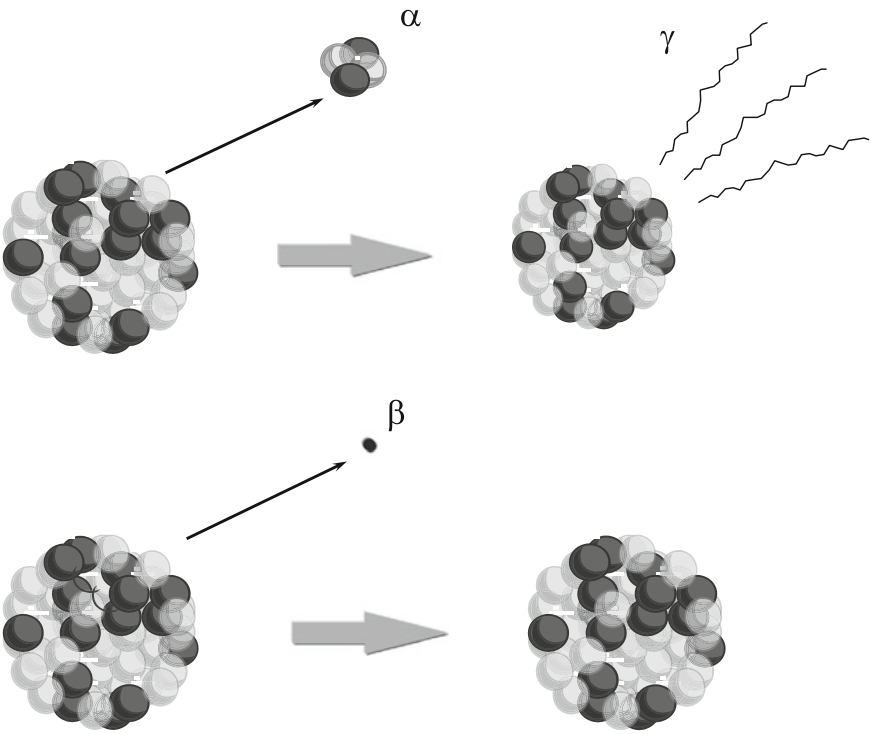


**Fig. 1.2** Radioactive decay by emitting $\alpha$ particles (mass number changes) or $\beta$ particles

Radioactivity is then the process of spontaneous decay by which a nucleus loses energy, emitting particles or electromagnetic radiation, or transforms itself into other nucleus with a different atomic number ($Z$). Emitted radiation energy is characteristic for each radionuclide. The type of emitted particle, its energy, and the time average of the emission will depend only on the nature of the radionuclide and not be altered by surrounding influences as pressure, temperature, electric or electromagnetic fields, or chemical reactions.

As shown in Fig. 1.2, unstable nuclei undergo spontaneous nuclear transformations either decreasing their mass number by ejecting two protons and two neutrons (alpha decay and spontaneous fission), or by transforming protons and neutrons into the other within the nucleus to emit beta radiation in the form of an electron or a positron[2] (beta decay), which alters the structure but keeps invariable the mass number. Gamma radiation occurs when the nucleus changes from a higher level energy state to a lower one, which often accompanies the spontaneous alpha and beta decay. A gamma spectrum is made up of discrete lines of energy quanta that range from a few keV to various MeV.

[2]It is a particle identical to an electron except that it has a positive electrical charge.

The time required for half of the nuclei of a specific radioisotope to undergo radioactive decay is referred to as radioactive half-life or simple half-life, denoted by the symbol $T½$. The inverse amount is the decay constant, a positive constant used to describe the rate of exponential decay. For different radionuclides, half-life varies from fraction of seconds, hours, and days, up to thousands of years. For example, radium-226, used for many years for cancer therapy, has a half-life of 1,620 years, while technetium-99m, used today for the diagnostic of diseases, has a $T½$ of 6 h.

Before Bohr and the discovery of radioactivity by Becquerel in 1896, the physicist Wilhelm Roentgen incidentally discovered a new and different kind of radiation, which he named *X*-rays (meaning unknown type of radiation). It was later established that *X*-rays are a special case of bremsstrahlung radiation, i.e., they are photons produced by the deceleration of high-speed charged particles when deflected by an electron by the atomic nucleus of a material elected as a target. When the electrons are suddenly stopped or deflected at the atoms, they release their energy in the form of electromagnetic radiation with a continuous spectrum. At the same time, within the excited atoms of the target, the electrons transit from an outer shell to an inner orbit releasing the difference between both energy levels in the form of an *X*-ray photon, which is characteristic for the excited atom.

*X*-rays spectrum, as shown in Fig. 1.3, is produced by two different mechanisms: bremsstrahlung and characteristic radiation. The energy peak of the spectrum corresponds to the maximum energy of accelerated electrons, e.g., if applying a voltage of 100 kV to an *X*-rays tube the electron maximum energy would be 100 keV. Electron energy emitted as *X*-rays increases with target atomic number and electron acceleration.

A simplified *X*-rays generator is illustrated on Fig. 1.4. The emission of electrons into a vacuum tube from an overheated cathode is collected on an anode. Applying a voltage between electrodes (tens or hundreds of kilovolts) allows the electron to be accelerated. Then, when the accelerated electrons hit the anode, it generates electromagnetic radiation (bremsstrahlung and characteristic *X*-rays) with an average energy proportional to the applied voltage.

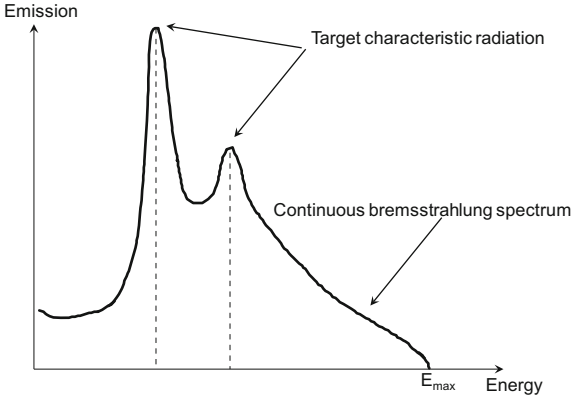


**Fig. 1.3** *X*-rays spectrum

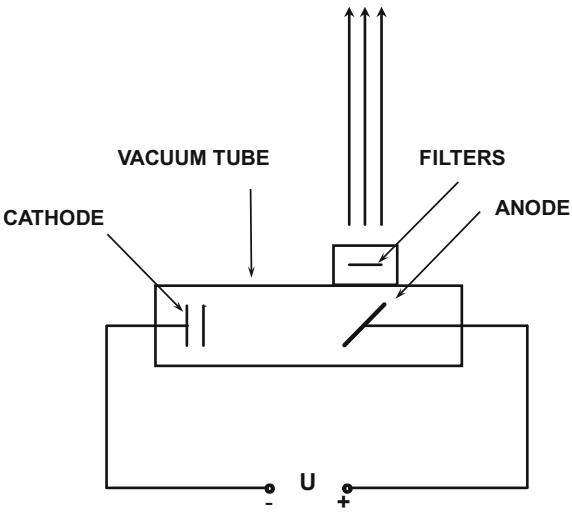


**Fig. 1.4** Diagram of an *X*-ray generator

Ionizing radiation is then produced by unstable nuclei, or by *X*-ray machines or other high-voltage devices. Ionizing radiation could be in the form of charged particles: energetic alpha ($\alpha$) particles or helium nuclei ($He^4$); beta ($\beta$) particles or high-speed electrons emitted by specific nuclides; neutral particles: neutrons obtained as a result of nuclear reactions; gamma ($\gamma$) radiation or electromagnetic radiation emitted by certain nuclides by radioactive decay; and *X*-rays: electromagnetic radiation from atomic electronic shells, with sufficient energy to remove electrons from molecules or atoms of the medium through which it passes, thereby producing ionization.

Ionizing radiation transfers part or the whole energy it carries to the atoms and molecules by different interaction processes which will depend on the character of the incident radiation, its energy, and the nature of the medium.

Energetic charged particles interact with matter by electrical (coulomb) forces causing ionization and excitation. They are known as direct ionizing radiation Excitation occurs when the interacting particle adds enough energy to an electron or a nucleon of an atom to move it to a higher energy level. After excitation, the excited atom will eventually lose its excess energy, releasing it as a photon of electromagnetic radiation. Ionization occurs when the radiation removes electrons from an atom causing negatively charged electrons to combine with positively charged nuclei, thus creating an ion-pair. The number of ion-pairs produced in a given track quantifies the amount of energy delivered by radiation.

Since charge particles continuously lose their energy by ionization and excitation while passing through matter, their path length is very limited. As represented in Fig. 1.5, due to the higher ionization capacity of alpha particles their maximum path length is scarcely various centimeters on air and some tens of micrometers in tissue. Alpha particles might be restrained by a sheet of paper. The maximum path length of beta radiation is some millimeters in tissue and a few meters on air.

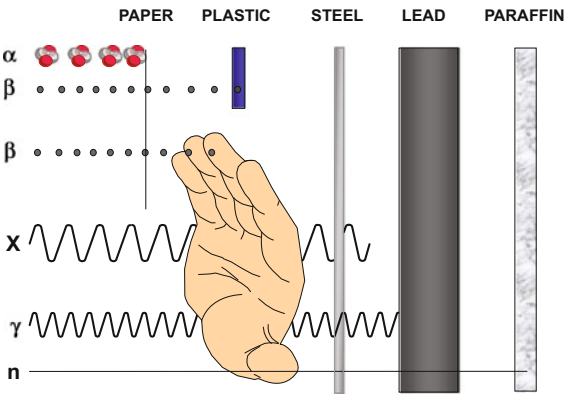


**Fig. 1.5** Penetrating properties of ionizing radiations

Non-charged particles, among them electromagnetic energy quanta (photons) and neutrons, are known as indirect ionizing radiation. They do not lose their energy continuously by ionization and excitation and, therefore, can travel large distances before colliding to medium's nuclei or electrons. The penetrating ability of radiation depends on the rate at which the radiation deposits energy along its path by collision. The probability of a collision increases with the traveled distance. Hence, indirectly ionizing radiation dissipates all or a major part of its energy in discrete interactions producing one or more charged particles known as "secondary particles", which in turn are responsible for ionization.

Both *X*-rays and gamma radiation are electromagnetic radiation. Absorption of electromagnetic radiation is more effective in materials of bigger atomic number ($Z$) and density, like lead and tungsten. Neutrons lose its energy more easily when traveling through materials of low atomic number ($Z$), preferable hydrogenated. It is remarkable that hydrogen is the only atom that can stop a fast neutron in a single collision.

The main processes of electromagnetic radiation interaction with matter are (a) photoelectric absorption; (b) Compton Effect; and (c) pair production. In the photoelectric (photon–electron) interaction, a photon transfers all its energy to an electron located in one of the outer atomic shells; the electron is then ejected from the atom and loses its energy, ionizing the medium in a relatively short distance. This interaction is possible only when a low energy photon (several tens or hundreds of keV, depending of the atomic number of the absorbing atoms) has sufficient energy to overcome the binding energy and remove the electron from the atom. A Compton interaction (scattering) is one in which only a portion of the energy is absorbed and a new photon is produced with reduced energy. This photon leaves the interaction site in a direction different from that of the original photon and the struck electron recoils at another angle in the forward direction. Compton scattering is the dominant mode from intermediate energies (0.1 MeV) to about the threshold for pair production (10 MeV). Pair production occurs when all photon energy is converted into mass in the vicinity of a nucleus and only with photons

with energies in excess of 1.022 MeV. In a pair-production interaction, the photon interacts with the nucleus producing a pair of particles, an electron and a positron.

Neutron interactions are more complex; they go through strong nuclear interactions and may occur almost exclusively by colliding with nuclei. The most important force to neutrons is the strong force which is responsible for binding the neutron's three quarks into a single particle. A neutron is stable when bound to a nucleus by the nuclear force. A free neutron can travel a considerable distance through many atoms without having a collision. It has an encounter only when it gets within the short range of the strong force of an atomic nucleus. It also decays in its free state with a time constant of $\sim 15$ min to a proton accompanied by the emission of a $\beta^-$ particle and an antineutrino to become a proton.

Neutron interactions with nuclei strongly depend on neutron energy, and are described in terms of quantities called cross sections and one of two primary types: scattering or absorption (capture). Scattering events can be subdivided into elastic and inelastic scattering. Elastic scattering occurs when a neutron interacts with a nucleus but does not leave the nucleus in an excited state, even though the neutron loses a fraction of its energy in each collision. In elastic collisions, part of the neutron's energy is transferred to nuclei in a manner analogous to a purely mechanical collision process.

The neutron may be absorbed or captured instead of being scattered by a nucleus. A variety of emissions may follow in this case. The nucleus may rearrange its internal structure, and release one or more gamma rays. Charged particles may also be emitted and the most common ones are protons, deuterons, and alpha particles. The nucleus may also rid itself of excess neutrons.

High-energy neutrons can also interact with nuclei by direct interaction. A direct collision between the neutron and the nucleus in this type of interaction results in the ejection of one or more nucleons (protons or neutrons) and the absorption of the incident neutron.

Table 1.1 summarizes the basic types of ionizing radiation and describes the most important peculiarities of their interaction with matter.

**Table 1.1** Types of ionizing radiation, examples, and peculiarities of interaction

| | | |
|---|---|---|
| *Direct ionizing radiation* | | |
| Charged particles | Alpha particles<br>Beta particles | 1. Continuous energy loss along their path<br>2. Dissipate its energy ionizing and exciting atoms on its path<br>3. Short path length |
| *Indirect ionizing radiation* | | |
| Non-charged particles (electromagnetic radiation) | Gamma radiation<br>*X*-rays | 4. Discrete energy loss lengthwise<br>5. Energy transfer to charge particles (and/or photons in case of neutrons) which will then ionize the medium<br>6. No specific path length (the likelihood of interaction increases with distance) |
| Non-charged particles | Neutrons | |

## Further Reading

1. Turner JE (2008) Atoms, radiation, and radiation protection, 3rd edn. Wiley
2. Attix FH (1991) Introduction to radiological physics and radiation dosimetry, 1st edn. Wiley
3. Nikjoo H, Uehara S, Emfietzoglou D (2012) Interaction of radiation with matter. CRC Press
4. Podgorsak EB (2005) Radiation oncology physics: a handbook for teachers and students. IAEA, Vienna
5. Cember H, Johnson T (2008) Introduction to health physics, 4th edn. McGraw-Hill, New York
6. Murray RL, Holbert KE (2014) Nuclear energy: an introduction to the concepts, systems, and applications of nuclear processes,7th edn. Elsevier
7. Johnson TE, Birky BK (2011) Health physics and radiological health, 4th edn. Lippincott Williams and Wilkins
8. Born M (1969) Atomic physics, 8th edn. Blackie and Son Limited, Glasgow

# Chapter 2
# Biological Effects of Ionizing Radiation

The study of the biological effects of ionizing radiation started practically at the same time as the discovery of *X*-rays in 1895. Since the techniques and methods accepted today to quantify radiation dose were absent at that time, first findings and studies were barely qualitative. Nonetheless, harmful effects to man and to laboratory animal species from ionizing radiation were already observed in the early years of the 20th century, when lack of data was a shared concern and there were no radiological protection standards. The first quantitative studies in experimental radiobiology developed during the 1920s, with the results of the epidemiological studies on the survivors of the atomic bombing of Hiroshima and Nagasaki, and data obtained from studies on patients exposed to radio therapeutic treatments, currently provide a large amount of information on the health effect of ionizing radiation which is the base of safety standards for occupationally exposed individuals.

Biological effects of ionizing radiation are a consequence of the ionization of atoms of biomolecules, which might cause chemical changes and alter or eradicate its functions. As illustrated in Fig. 2.1, energy transmitted by radiation may act directly causing ionization of the biological molecule or may act indirectly through the free radicals resulting from the ionization of the water molecules that surround the cell.

Due to ionization, proteins can lose the functionality of its amino groups and modify its behavior, thus increasing its chemical responsiveness; enzymes would be deactivated; lipids will suffer peroxidation; carbohydrates will dissociate; and nucleic acids chains will experiment ruptures and modifications of structure. But from all possible combined alterations, DNA is the primary target for radiation because it contains genes/chromosomes that hold information for cell functioning and reproduction that are critical to cell survival.



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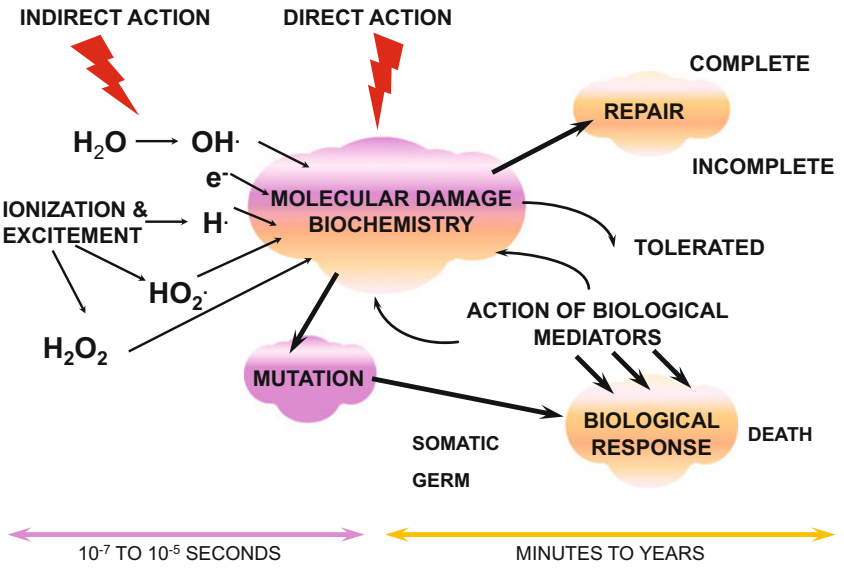


**Fig. 2.1** Physical, biochemical and biological response mechanisms

As a result of radiolytic decomposition of water by ionization and excitation, hydrogen, and hydroxyl radicals could combine to form toxic substances as hydrogen peroxide ($H_2O_2$), which can also contribute to the destruction of cells.

The deposition of energy by ionizing radiation is a random process. Even at very low doses there is some probability that enough energy may be deposited into a critical volume within a cell to result in cellular changes or cell death. But thanks to the remarkable ability of cells to repair damage, enzymatic, and repair mechanisms would lead in many instances to the correct DNA repair and the cell will survive without any modification to its function or genetic structure. If the repair of DNA damage is incomplete, signaling pathways[1] leading to cell death through apoptosis, terminal differentiation, and senescence are activated. Physical processes of energy absorption and induced ionization and excitation, as well as biochemical processes triggered by the living organism response, would occur within fraction of seconds. Repair of cellular damage, such as DNA repair, may take from minutes to hours after exposure depending on the type of damage.

Another possible result is mutation. The cell will survive but with modification in the DNA sequence of the cell's genome. Mutated cells are capable of reproduction and thus perpetuate the mutation. If the mutated cell is a somatic cell,

[1]A signaling pathway describes a group of molecules in a cell that work together to control one or more cell functions, such as cell division or cell death. After the first molecule in a pathway receives a signal, it activates another molecule. This process is repeated until the last molecule is activated and the cell function is carried out.

mutation could lead to a malignant tumor. If the mutated cell is a germ cell, it may cause a hereditary effect. These are stochastic effects and their consequences (cancer or hereditary effects) may be statistically observed long after exposure.

If damage cannot be repaired cell death occurs. Cell death means the loss of a specific function for differentiated cells which do not replicate, such as nervous cells, muscle cells, or secretory cells. For proliferating cells, such as primary blood-forming (hematopoietic) cells or cell growing in a culture media (stem cells), cell death means the permanent loss of their proliferating capacity or the loss of their reproductive integrity. If many cells die, there will be tissue and organ damages which may cause a rapid, whole body response. Figure 2.2 shows both paths by which radiation may affect the whole body system.

Cellular sensitivity to radiation has been better studied in proliferating cells. For proliferating cells, radiosensitivity depends on a number of factors of which the most important are cell proliferation capacity; cell differentiation degree; phase of the cell cycle at which the irradiation occurs; radiation quality; dose rate; and dose fractioning. In general, cellular sensitivity to radiation is directly proportional to the rate of cell division and inversely proportional to the degree of cell differentiation (Law of Bergonie and Tribondeau 1906). This explains why tissues with a high turnover rate are more radiosensitive than those that do not have a continuously turnover. Related to the cell cycle, cells are more sensitive to radiation during mitosis (cell division) than through the preceding substages when the cells are not

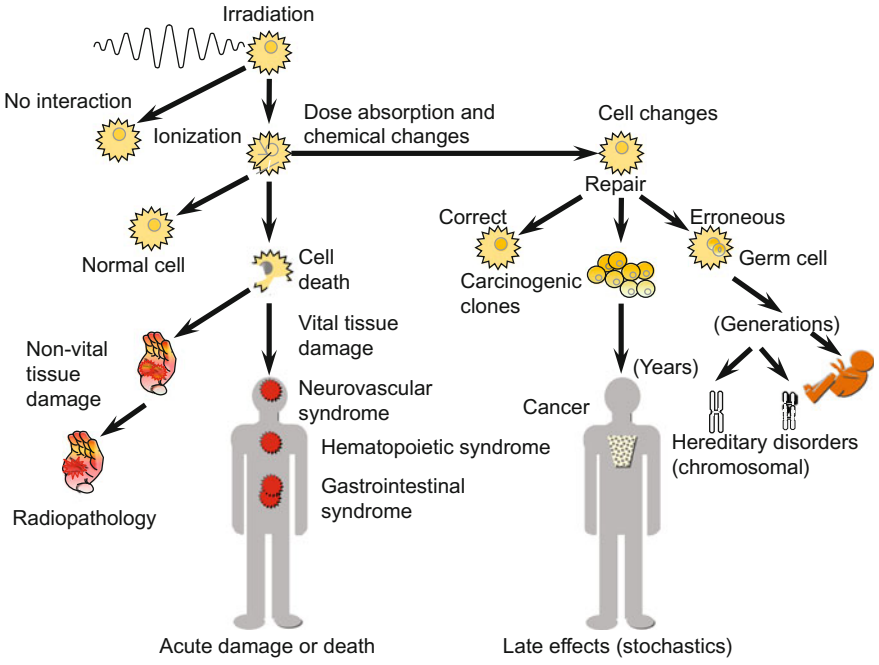


**Fig. 2.2** Radiation effects on the whole body system

dividing and the mechanisms of repair (cell cycle checkpoints[2]) are activated. Some cells, like nerve cells, do not undergo much division. Most cells have a moderate cell rate division. For human organism, it might be considered that lymphocytes, stem cells in the bone marrow, cells of the lens of the eye, and epithelial cells of gastrointestinal tract are the most radiosensitive; surface of the stomach walls, esophagus, mouth, and skin are moderate radiosensitive; while muscle cells, bone cells and nerve cells are low radiosensitive.

Ionizing radiation is more effective at producing biological damage when its LET (linear energy transfer) is high, the dose rate is high and the period of time between consecutive exposures is short.

## 2.1 Classification of Biological Effects

Biological mechanisms can act in favor of tissues to maintain its functionality with a loss of certain number of cells. But when the radiation damage is of such magnitude that the cell killing cannot be compensated by the cellular turnover, tissue functionality is not possible further, leading ultimately to acute organ damages or death. Another concern is the role radiation induced mutations have in carcinogenesis. The risks of cancer after high and moderate doses of radiation are relatively well understood from detailed epidemiological studies of the Japanese atomic bombing survivors and others. Although only down to about 100 mGy[3] risk of mortality and morbidity is proportional to radiation dose [1].

According to the last ICRP recommendations [2], adverse health effects from radiation exposure are grouped in two general categories, i.e., harmful tissue reactions (deterministic effects) and stochastic effects (of random or statistical nature).

Harmful tissue reactions (deterministic effects) resulting from the killing/malfunctioning of cells is characterized by a certain dose called "threshold." The reason for the threshold is that a serious malfunction or death of a critical population of cells in a given tissue should be sustained before injury is expressed in a clinically relevant form. As shown in Fig. 2.3, the frequency of the injury increases with dose as the number of affected cells is directly proportional to the severity of the effect. The graphic on the right in Fig. 2.3 illustrates the way in what the severity of the damage above the dose-threshold, including the impairment of the capacity for tissue recovery, increases with dose. It has been presently recognized that tissue reactions can be modified after irradiation by the use of some biological response modifiers. Some examples are antioxidants, radical scavengers, apoptosis inhibitors, anti-inflammatory drugs, growth factors, etc.

[2]Cell cycle checkpoints are control mechanisms which ensure proper cell division. Each checkpoint serves as a potential halting point along the cell cycle, during which the conditions of the cell are assessed, with progression through the various phases of the cell cycle occurring when favorable conditions are met.

[3]Gy (gray) is the unit of absorbed dose used in radiation biology, clinical radiology, and radiation safety. It describes the energy imparted to matter by all kinds of ionizing radiation.

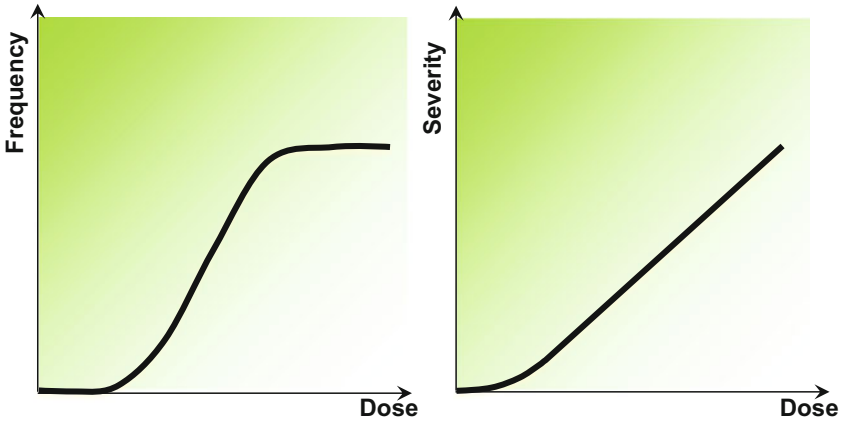


**Fig. 2.3** Relationship between dose and the frequency or severity of tissue reactions

Tissue reactions are also characterized by different periods of latency, so it could be distinguished between early tissue reactions detected in a few days or weeks (on a timescale of hours to weeks), and late tissue reactions, detected months to years after the irradiation. Early tissue reactions may be of the inflammatory type, resulting from cell permeability changes and histamine release (e.g., erythema), or they may be reactions resulting from cell loss (e.g., mucositis, and desquamatory reactions in epithelial tissues) [1]. Late tissue reactions are called "generic" if they arise as a direct result of damage to that tissue, for instance a vascular occlusions leading to deep tissue necrosis after protracted irradiations or "consequential" if they occur as a result of an early cellular damage, e.g., a dermal necrosis as a result of severe epidermal denudation and chronic infection [3].

In view of different individual radiosensitivity, it is accepted that the dose-threshold for a specific tissue reaction is the dose that produces the same tissue reaction in the 1–5 % of the total exposed individuals. Updated information on dose thresholds corresponding to doses that result in about 1 % incidence of morbidity and mortality for various organs and tissues is available in the ICRP 2007 Recommendations [2]. Some examples are temporary sterility in 3–9 weeks from an acute absorbed dose of $\sim$0.1 Gy in the testes; depression of blood forming process in 3–7 days from an acute absorbed dose of $\sim$0.5 Gy in the bone marrow; and cataracts (visual impairment) in several years from an acute absorbed dose of $\sim$0.5 Gy [3]. Ordinarily, fractionated doses or protracted doses at low dose rates are less damaging than acute absorbed doses.

The most severe tissue reaction is death. Mortality after irradiation is generally the result of severe cell depletion in tissues or other major dysfunction of one or more vital organs of the body. Enough high acute doses to the whole body in very short periods of time may lead to lethal disorders. Although there is great uncertainty in the lethal dose-threshold on account of the general health of individuals, the immediate medical assistance received and other specific factors, absorbed doses

between 3–5 Gy may cause death in 50 % of exposed individuals in a lapse of 1–2 months. ICRP 2007 Recommendations indicate a $LD_{50/60}$, i.e., within 60 days, for a normal human healthy adult of around of 4 Gy midline dose [2]. In between 6–10 Gy, the mortality % increases and the time period to the death decreases. In this dose range (3–10 Gy) the cause of death is hematopoietic failure, i.e., hematopoietic syndrome, primarily from a lack of bone marrow progenitor cells, as well as from hemorrhages without the replacement of radioresistant red cells. Acute doses approaching 10 Gy causes severe gastrointestinal damage (see Fig. 2.2), which when combined with hematopoietic damage causes death in 1–2 weeks with little possibilities of survival. Local acute doses above 10 Gy to the lungs produce an acute inflammation (pneumonitis) that may lead to death. Renal damage occurs in the same dose range if the kidneys are irradiated. Neurovascular syndrome appears at even higher doses toward 50 Gy and above, and induces the death in 48 h [4].

The main concern of radiation safety at low doses has been radiation induced cancer and hereditary diseases, meaning by low doses ~100 mGy and less. In the very early days of radiation protection standards it was assumed that "genetic damage" from radiation (meaning hereditary effect), would accumulate across generations and eventually have a marked impact on the health of human populations.

Since recommendations adopted in the late 1970s (ICRP 26) until present, it has been assumed that, stochastic or probabilistic effects may occur at low doses, and are generally considered to be cancers (including leukemia) and genetic defects in the progeny. This assumption has implied that there is a linear no-threshold increase in genetic cell damage as a function of radiation dose, and that each unit of radiation would increase the risk. This approach is consistent with the so-called linear, no-threshold (LNT) hypothesis, accepted because it is still not actually known today what level of risk is associated with very low-dose exposure to radiation. LNT hypothesis was considered to be a prudent judgment for public policy aimed at avoiding unnecessary risk from exposure.

However, in this half-century a lot of research has been done and many advances in modern molecular biology and instrumentation have taken place. As explained before, in the current conventional interpretation of radiation carcinogenesis, ionizing radiation acts primarily by damaging nuclear DNA (much of which is repaired by repair systems), and inducing targeted DNA mutations in stem cells thus initiating the cancer development process. Secondary mutations ultimately accumulate, leading to a malignant neoplasm development [5]. ["From Biological Mechanisms of Radiation Actions at Low Doses. A white paper to guide the Scientific Committee's future programme of work, by UNSCEAR, ©2012 United Nations. Reprinted with the permission of the United Nations".].

Now it is recognized that acute doses or doses experienced in a protracted form of up to ~100 mGy (low LET or high LET), produce no tissue functional impairment [2]. Besides, according to more recent advances in studies of biological effects of ionizing radiation, it is evident that there are much more data describing how biological systems respond to low doses of radiation.

The major areas of study are related to three unique biological responses: bystander effects, adaptive responses, and genomic instability [5]. Because of bystander effects, tissues respond as a whole to ionizing radiation and not as single cells; they demonstrate that even though the energy is deposited in random defined sites, radiation effects are not limited to the individual cells where the energy is deposited. Bystander effects at low doses of low-LET radiation appear to induce biochemical and functional cell and tissue responses that express both damage and adaptive cell protection. Also, a consensus is emerging that low-LET irradiation below 0.5 Gy does not cause transmissible genomic instability. Genomic instability suggests the accumulation of multiple changes to convert a stable genome of a normal cell to an unstable genome characteristic of a tumor, and recent reviews indicate and confirm a likely threshold for the induction of such transmissible instability.

Protective response categories involve cellular defenses, such as radical detoxification; cell removal by immune response—cells with altered phenotype may be detected and killed by the immune system—and cell removal through intracellular signaling, such as by cell differentiation and apoptosis. It has also been suggested the existence of a protective adaptive response [6]. Adaptive response is defined as the temporary modulation (usually reduction) by prior small doses of the response to subsequent high radiation doses [5] ["From Biological Mechanisms of Radiation Actions at Low Doses. A white paper to guide the Scientific Committee's future programme of work, by UNSCEAR, ©2012 United Nations. Reprinted with the permission of the United Nations".].

Supported by the fact that uncertainties regarding the role of these processes in cancer risk are currently too great for the development of practical judgments, the Commission recommended again that the practical system of radiological protection continue to be based upon the assumption that at doses below about 100 mSv[4] a given increment in dose will produce a directly proportionate increment in the probability of incurring cancer or heritable effects attributable to radiation [2].

Relation between dose and risk based on the conservative approach and LNT hypothesis is shown in Fig. 2.4. This relation implies that when the dose increases the risk of late health effects like cancer, noncancer and heritable diseases increases, but in the absence of other modifying factors, the severity of the effect is not expected to increase.

Health effects of low doses of radiation continue to be a concern and are a priority so as to take its result into official standards. A bill was passed recently to revitalize the existing DOE low-dose radiation research program so as to increase the understanding of the health effects of low doses of ionizing radiation. The bill calls for a study by the National Academies and seemingly a new Biological Effects of Ionizing Radiation (BEIR) report—the BEIR VIII report—is on its way.

[4]Sv (Sievert) is a special unit for the quantities equivalent dose, effective dose, and operational dose, used in radiation safety to reflect the amount of radiation detriment likely to result from the dose, or the amount of harm caused by the dose to a tissue or organ.

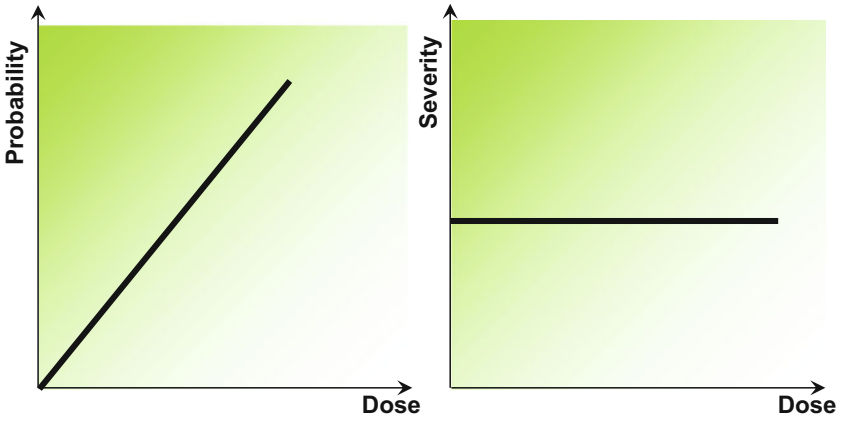


**Fig. 2.4** Relationship between probability and severity of stochastic effects with dose

Heritable diseases mean induction of cancers from irradiated somatic cells and genetic diseases in offspring following parental germ cell irradiation. They are not to be confused with health effects resulting from an exposure during prenatal development. Even though there is no direct evidence of heritable effects to humans, experimental observations in animal and plant species exposed to relatively high doses, give good reasons to include such risk for future generations as part of the system of protection. However, the BEIR VII report [7] concluded that with low dose or chronic exposures to low-LET irradiation, the risk of adverse heritable health effects to children conceived after their parents have been exposed is very small compared to baseline frequencies of genetic diseases in the population.

## 2.2 Radiation Effects During Prenatal Development

Prenatal development is the process in which an embryo or fetus gestates during pregnancy. As cells are rapidly dividing and forming the new tissues and organs, the embryo/fetus is considered to be at the most radiosensitive stage of human development, particularly in the first 20 weeks of pregnancy. The effects of radiation exposure during prenatal development depend on the time period when the exposure occurred. Time period of pregnancy is estimated in days or weeks after last menstrual cycle and reflects the developmental stage of the embryo/fetus.

During the first week after the menstrual cycle, radiation exposure to the uterus is not dangerous provided that ovum fertilization occurs about the 14th day after. The embryonic period in humans begins at the moment of fertilization and continues until the 20th day (about 3 weeks). The only effect that could be expected in this period is failure to implant. Typically, it is estimated that 30–50 % of pregnancies are lost spontaneously precisely during this period.

From the study of atomic bomb survivors, it was learned that between 3 and 8 weeks of pregnancy the more common expected effect after irradiation is the induction of malformations. A true dose-threshold of around 0.1 Gy has already been considered for this effect, so risk of malformations is not expected after an in-uterus exposure to doses well below 100 mGy.

Japanese atomic bomb data on the induction of severe mental retardation (SMR) after irradiation at the most sensitive prenatal period (weeks 8–15) support a true dose-threshold of 0.3 Gy for this effect, and the total absence of risk at lower doses.

The major concern after 15 weeks of pregnancy is the cancer risk following in-uterus irradiation provided that some evidences suggest an excess of cancers and leukemia. Indeed, the largest studies of in-uterus medical irradiation found an increment of all types of childhood cancer by approximately the same degree, with a relative larger risk for leukemia than for solid tumors [8]. Early indications are that, between the 16th week of pregnancy and birth, cancer risk from prenatal radiation exposure is similar to, or slightly higher ($\sim$2 %) than, the normal expected cancer risk from exposure in childhood (which is 48–50 %). The increased risks will depend on the amount of radiation to which the baby was exposed and the amount of time that it was exposed.

Termination of pregnancy owing to radiation exposure is an individual decision affected by many factors. Nevertheless, absorbed doses below 100 mGy to the embryo/fetus should not be considered a reason for terminating a pregnancy.

## 2.3 Biological Indicators of Radiation Damage

Even though safety is assured through a conscious fulfillment of requirements and regulations, the potentiality of an accident could not be excluded. In events like an accident, it is possible that some workers and, occasionally, members of the public would be receiving doses above the established limits, i.e., experience overexposure. If an overexposure occurs, biological indicators play an important role in defining the severity of the damage, to estimate the exposure retrospectively in some cases, and to help anticipate the occurrence of late effects. They can improve the diagnosis and treatment of injured individuals as well.

There are two main types of biological indicators, i.e., clinical indicators and laboratory test indicators. Clinical indicators of acute radiation damage are categorized according to damaged organs/tissues and symptoms in three different subgroups hematopoietic, gastrointestinal, and neurovascular.

Hematopoietic and gastrointestinal symptoms in advance of acute radiation disease are nausea, vomiting and anorexia within a few hours at the higher dose levels or after 6–12 h at the lower dose levels. Neurovascular indicators of acute radiation disease are severe tiredness (weakness/fatigue), apathy, disinterest, sweating, fever, headache, and ataxia.

As shown in Table 2.1, the severity of an overexposure (and thence the probable dose) might be predicted by the time elapsed from the exposure and onset of

**Table 2.1** Prognosis of acute radiation damage by clinical indicators [9]

| Dose (Gy) | Initial symptoms | | Critical period | Post exposure prognosis | Lethality | |
|---|---|---|---|---|---|---|
| | % Incidence | Time until onset | | | Percent | Time after exposure |
| >50 | 100 | minutes | 1–48 h | Hopeless | 100 | 1–48 h |
| 10–15 | 100 | 30 min | 5–14 days | Really bad | 90–100 | 2 weeks |
| 5–10 | 100 | 0.5–1 h | 2–6 weeks | Poor | 0–90 | Weeks or months |
| 2–5 | 50–90 | 1–2 h | 2–6 weeks | Poor | 0–90 | Weeks or months |
| 1–2 | 0–50 | >5 h | – | Excellent | – | – |
| 0–1 | 0–10 | | – | Excellent | – | – |

symptoms. Generally, as more stable the symptom and lesser the elapsed time, the higher the dose and life-threatening to the injured person.

At about 3 Gy or over, depending on the particular radiation energies involved, general symptoms and signs also include erythema (reddening of the skin due to inflammation) within hours or days, and epilation (removal of hair) after about 2 to 3 weeks, which is a confirmatory finding.

Laboratory test indicators are cytogenetic, hematological and biochemical indicators.

Cytogenetic indicators—also known as cytogenetic dosimetry or biological dosimetry based on cytogenetic methods—are currently the more effective biological dosimeter. They have eventually become a routine component of radiation safety programs. From four possible cytogenetic methods currently available, the most consistent method is based on the significant increase of the frequency of chromosomal aberrations, in particular dicentrics, in peripheral blood lymphocytes as a result of irradiation. Relevant calibration curves for aberrations can be obtained using human lymphocytes in vitro, then, it is possible to use the frequency of aberrations measured in lymphocytes to estimate radiation dose. A dicentric aberration is an unstable aberration whose frequency decreases with time after exposure. The dicentric aberration frequency depends on cell turnover rate and can be relatively long in nonproliferating cells. The background concentration of dicentric cells in unirradiated persons is low, as little as 1–2 dicentrics per 1,000 cells in T-lymphocytes, and there is little variability among individuals, so that small radiation-induced increases can be quantified. It must be kept in mind that the frequency of aberrations decrease with time, so this method is useful only for individuals recently exposed to radiation. This indicator can be used to estimate doses as low as 0.1 Gy, as well as to establish the homogeneity of exposure and if it is not homogeneous, estimate the part of the body irradiated. Below 0.1 Gy, the sample size required for statistically reliable results is so large that it is impractical to obtain [10].

Methods for estimating radiation dose using biological indicators have made rapid progress during the recent years. While dicentric chromosome method still plays a central role, it is no longer the only quantitative approach in biological dosimetry. Evaluation of the frequency of stable chromosome aberrations (those

that do not decrease with time) has been made possible by techniques that measure translocations between chromosomes. This is done by evaluating banded chromosome preparations or by using a less accurate but more rapid size-grouping method. Such techniques were useful in measuring aberrations in the survivors of the atomic bombing of Hiroshima and Nagasaki at long times after radiation exposure occurred [10, 11].

A promising cytogenetic method—particularly when receiving protracted exposures or if the exposure occurred a long time ago—is the "fluorescent in situ hybridization" (FISH) (commonly called the chromosome painting technique), which can be used to further define stable chromosome aberrations and obtain full genomic translocation frequencies. Since this method allows scoring both dicentrics and translocations in the same cell, it is considered of improved efficiency for detecting exchange aberrations and for enabling a higher confidence in estimating radiation dose [12]. Theoretically, it could detect lower doses than is possible using conventional cytogenetic methods.

Further cytogenetic indicators in process of study include MN (micronuclei) analysis and premature chromosome condensation (PCC) analysis [11]. The first is based on the measurement of micronuclei in populations of exposed cells. Micronuclei are formed when cells with broken chromosomes divide; the evaluation of micronuclei is then much easier to perform than is of chromosome. To ensure that only dividing cells are scored, cells are treated with cytochalasin B, which blocks cytokinesis—a process whereby the cytoplasm of a single cell is divided to spawn two daughter cells—and results in binucleated cells. Only the binucleated cells are evaluated for the formation of micronuclei. The PCC analysis consist of the fusion of human lymphocytes with Chinese hamster ovary mitotic cells in the presence of a fusing agent, polyethylene glycol, to enable the measurement of chromosomal aberrations immediately following irradiation without the perturbing influence of processes associated with cell cycle progression to mitosis (cell division) [11].

A common hematological indicator as peripheral blood count can be especially valuable for an early diagnostic and prognosis of the severity of the damage after exposure; it might also serve to obtain a gross dose estimate. Hematopoietic system is known to be one of the most radiosensitive and the reason why blood cell count has maintained a time-honored position as a screening indicator for various disease states. From 0.5–1 Gy it will be well-observed identified changes in the peripheral blood cell counts, like an increase in neutrophil count, severe lymphopenia, and a decrease in the platelet count (thrombocytopenia). Lymphocytes are especially sensitive to radiation, and they may succumb to interphase death after exposure to a dose of only 0.05–0.15 Gy.

There are also some biochemical indicators that play an important role in diagnostic and prognosis of acute radiation damage. Conventional biochemical indicators can be used based on the fluctuation of certain metabolites in human urine and blood after overexposure. They include creatine/creatinine, taurine, and alpha amylase, as well as ALAT (serum alanine aminotransferase), ASAT (aspartate aminotransferase), and γGT ([gamma]-glutamyl transpeptidase), but none of

these parameters is specific to radiation-induced damage with the exception of alpha amylase in serum, which is specific for radiation damage to the parotid glands.

Management of more recent accidents has demonstrated that there are two more biological indicators specific to certain vital importance organs. They are blood citrulline level as an indicator of radiation damage to the intestinal epithelium [13], and Flt3 blood ligand concentration as an indicator of radiation damage to the bone marrow [14]. Flt3 ligand is a ligand for the FLT3 tyrosine kinase receptor and belongs to a small group of growth factors that regulate proliferation of early hematopoietic cells.

These indicators are not extremely practical as biological dosimeters but they offer the physician valuable qualitative information on the condition of the injured individual. In addition, when the dose received is distributed heterogeneously at organ and organism level, the knowledge of the global irradiation dose is important but not sufficient by itself.

## References

1. ICRP Committee 1 Task Group Report: C1 Foundation Document 11/109/05 (2005) Biological and Epidemiological Information on Health Risks Attributable to Ionizing Radiation: A Summary of Judgements for the Purposes of Radiological Protection of Humans
2. ICRP Publication 103 (2007) The 2007 recommendations of the international commission on radiological protection. Ann. ICRP 37(2–4)
3. ICRP Publication 118 (2012) ICRP statement on tissue reactions/early and late effects of radiation in normal tissues and organs—threshold doses for tissue reactions in a radiation protection context. Ann. ICRP 41(1–2)
4. Bertho JM, Griffiths NM, Gourmelon P (2004) The medical diagnosis and treatment of radiation overexposed people. Institut de Radioprotection et de Sûreté Nucléaire. http://w3.tue.nl/fileadmin/sbd/Documenten/IRPA_refresher_courses/The_medical_diagnosis_and_treatment_of_radiation_overexposed_people.pdf
5. UNSCEAR (2012) Biological mechanisms of radiation actions at low doses. In: A white paper to guide the scientific committee's future programme of work. United Nations, New York
6. Brooks AL (2012) A History of the United States Department of Energy (DOE) Low Dose Radiation Research Program: 1998–2008, Review Draft
7. The National Academies (2005), BEIR VII: Health Risks from Exposure to Low Levels of Ionizing Radiation. Report in Brief prepared by the National Research Council bases on the committee's report
8. ICRP Publication 90 (2003) Biological effects after prenatal irradiation (Embryo and Fetus). Ann. ICRP 33(2)
9. García Lima O (1998) Capítulo II Efectos biológicos de las radiaciones ionizantes. de Protección Radiológica en la Aplicación de las Técnicas Nucleares
10. Radiation Dose Reconstruction for Epidemiologic Uses (1995) The National Academies Press Open Book. http://www.nap.edu/openbook.php?record_id=4760&page=R1
11. International Atomic Energy (2001) Cytogenetic Analysis for Radiation Dose Assessment, Technical Report Series No. 405. IAEA, Vienna
12. Natarajan AT, e. alia (1996) Current Cytogenetic Methods for Detecting Exposure and Effects of Mutagenens and Carcinogens. Environmental Health Perspectives v. 104 Suppl. 3. http://www.ncbi.nlm.nih.gov/pmc/articles/PMC1469636/pdf/envhper00346-0033.pdf

13. Lutgens L, Lambin P (2007) Biomarkers for radiation-induced small bowel epithelial damage: an emerging role for plasma citrulline. World J Gastroenterol 13(22):3033–3042
14. Bertho JM, Griffiths N, Gourmelon P (2015) The Medical Diagnosis and Treatment of Radiation Overexposed People. *Downloaded from ResearchGate,* 24 October 2015

# Chapter 3
# Radiation Sources: Benefits and Risks

*X*-rays tubes and radium found immediately application in medicine from the discovery of radioactivity at the beginning of the 20th century. Since then, the inventory and amount of available radionuclides have significantly grown with the introduction of nuclear reactors and linear accelerators. Countless advances in nuclear science and technology helped to rapidly spread and increase the use of radiation sources[1] in most human activities. In today's world, it is difficult to envision a further development in medicine, industry, agriculture, education, and scientific research without them.

## 3.1 Radiation Sources in Medicine

The use of *X*-rays started shortly after their discovery. The first medically relevant radiographs were produced as early as 1896, almost at the same time when the use of *X*-ray for skin radiotherapy began. Radioactive radium also found application for cancer treatment from the very beginning of its full isolation, and very soon substituted *X*-rays for the treatment of skin cancer. $^{226}$Ra is an alpha emitter with a long half-life (1,600 years) used until the 1970s in the treatment of malignant tumors, particularly brachytherapy, intracavitary applications, and interstitial applications. *X*-ray generators are now used in medicine as therapeutic installations, ancillary radiotherapeutic installations, and medical imaging installations. State-of-the-art computed tomography (CT) is a medical imaging method, based on a large series of

[1]A source is defined as: Anything that may cause radiation exposure, such as by emitting ionizing radiation or by releasing radioactive substances or materials. For example, materials emitting radon are sources in the environment; a sterilization gamma irradiation unit is a source for the practice of radiation preservation of food; an *X* ray unit may be a source for the practice of radiodiagnostic; a nuclear power plant is part of the practice of generating electricity by nuclear fission, and may be regarded as a source (i.e., with respect to discharges to the environment) or as a collection of sources (i.e., for occupational radiation protection purposes).



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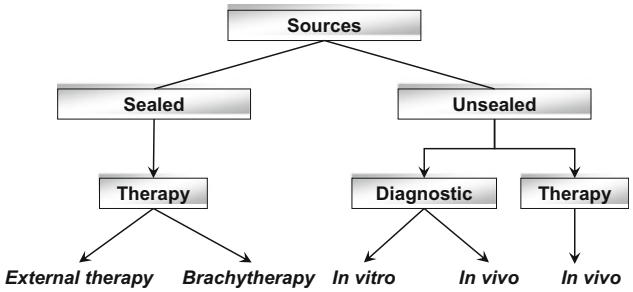



**Fig. 3.1** Radiation sources in medicine

two-dimensional *X*-ray images taken around a single axis of rotation and digitally processed, to obtain a three-dimensional image of the inside the body.

Current use of radiation sources in medicine is classified in two large groups depending on the purpose for which they are used—therapeutic and diagnostic. Radiation therapy uses high-energy radiation to shrink tumors and kill cancer cells. Diagnostic use, apart from *X*-ray imaging techniques, also implies the use of radionuclides to detect the presence of a disease and the extent to which it has invaded the body. Many diagnostic techniques are based on tracer and imaging techniques, i.e., labeling molecules of interest with radionuclides to study certain functional and biochemical processes that occurs in the body. Use of radiation sources in medicine is represented in Fig. 3.1.

Sealed sources[2] are used in brachytherapy[3] and external therapy applications. Beta emitters like $^{90}$Sr are currently preferred for contact therapy (surface brachytherapy) and ophthalmic applications, where they have totally replaced radium sources. Besides, there are special proceedings using permanent implants of $^{125}$I and $^{203}$Pd, e.g., for prostate cancer, which feature a relatively short half-life as its most important advantage.

Remote afterloading brachytherapy with low activity $^{137}$Cs or $^{192}$Ir sources, or high activity $^{192}$Ir sources ($\leq$ 0.4 TBq), have proven to be very useful to treat endometrial, cervical, prostate, or pancreatic cancer against radium sources. Remote afterloading brachytherapy uses a special machine which contains the radioactive sources. Plastic catheters (applicators) are placed previously in the patient, either interstitially or intracavitarily, and then the sources are introduced remotely into the hollow applicator by using a control mechanism through wires or compressed air.

Current radiotherapy divisions usually rely on $^{60}$Co teletherapy machines with activities of various TBq and electron/photon accelerators that have proven to be very efficient and reliable for external therapy of cancer. Some typical applications and sources are shown in Table 3.1.

[2]Sealed source is a radioactive material that is (a) permanently sealed in a capsule or (b) closely bonded and in a solid form.

[3]It is an internal radiation treatment or radiation therapy delivered from a short distance.

**Table 3.1** Some sealed sources used in medicine

| Application | Radionuclide | T½ | Source activity | Remarks |
|---|---|---|---|---|
| Bone densitometry | $^{241}$Am | 433 y | 1–10 GBq | Mobil units |
| | $^{153}$Gd | 242 d | 1–40 GBq | |
| | $^{125}$I | 60 d | 1–10 GBq | |
| Manual brachytherapy | $^{137}$Cs | 30 y | 50–500 MBq | Small portable sources |
| | $^{226}$Ra | 1,600 y | 30–300 MBq | |
| | $^{60}$Co | 5.3 y | 50–500 MBq | |
| | $^{90}$Sr | 29 y | 50–1500 MBq | |
| | $^{103}$Pd | 17 d | 50–1500 MBq | |
| | $^{125}$I | 60 d | 50–1500 MBq | |
| | $^{192}$Ir | 74 d | 200–1500 MBq | |
| | $^{131}$I | 8 d | 50–1500 MBq | |
| | $^{196}$Au | 2.7 d | 50–1500 MBq | |
| | $^{242}$Cf | 2.6 a | 50–1500 kBq | |
| Remote afterloading brachytherapy | $^{60}$Co | 5.3 a | ≈10 GBq | Mobil units |
| | $^{137}$Cs | 30 a | 0.03–10 MBq | |
| | $^{192}$Ir | 74 d | ≈400 GBq | |
| External therapy | $^{60}$Co | 5.3 y | 50–1000 TBq | Fixed installations |
| Blood sterilization | $^{137}$Cs | 30 y | 2–100 TBq | Fixed installations |

Neutron and neutron-capture therapy are rather effective methods for treatment of some human cancers that show promising results, however, these methods are still on trial in very expensive research facilities.

Stereotactic radiosurgery or stereotaxy—also called gamma knife—is an application which has been used as a very promising minimally invasive neurosurgery for brain tumors. This method uses a three-dimensional coordinates system to locate small targets and several gamma $^{60}$Co sources with an activity of 1 TBq or less in the form of tiny needles or arcs that deliver narrow, well-defined beams that conform to the lesions.

Other common application of sealed sources in medicine is radiosterilization of tissues for grafts, blood, pharmaceuticals, and disposable dental and medical supplies, such as plastic syringes, surgical gloves, suturing materials, and catheters.

Nuclear medicine uses unsealed[4] radioactive sources in the form of radiopharmaceuticals and imaging techniques for diagnostic, therapy and biomedical research. Nuclear medicine imaging differs from traditional imaging—computed tomography (CT) and magnetic resonance imaging (MRI)—in that it allows to see how the organ being investigated is functioning and to measure its chemical and biological processes. Radiopharmaceuticals and molecular imaging currently

[4]An unsealed source is a source that does not meet the definition of a sealed source.

support research at molecular and cellular level to better understand human diseases and develop more effective treatments.

In vivo studies imply the patient administration of radiopharmaceuticals to explore biochemistry and to obtain the most accurate diagnostic information with the lowest dose. The effective half-life of the radionuclide must be such that its retention in the organism was the minimum necessary to lead to the needed information. Radionuclide's emission energy must also be low, but will depend on the detection system. Radiopharmaceuticals, in conjunction with imaging techniques, such as gamma camera imaging, single-photon emission computed tomography (SPECT), and positron emission tomography (PET), have the ability to detect cancerous involvement often before symptoms occur.

From all low energy gamma emitters used in medical functional imaging, anthropogenic element technetium is the most explored metal ion for its complexation behavior. $^{99m}$Tc is a metastable nuclear isomer of $^{99}$Tc. It is a radionuclide with a half-life of 6 h and a gamma-ray emission of 140 keV of readily detectable energy.

Most commonly used radionuclides in SPECT imaging are $^{67}$Ga, $^{131}$I, $^{201}$Tl, and $^{111}$In. Radionuclides used in PET imaging are typically produced in a cyclotron. Among them, the most advantageous is $^{18}$F, particularly in the form of fludeoxyglucose F18 ($^{18}$FDG) initially developed for studying glucose metabolism in vivo, which today is the most useful clinical PET tracer for the detection, staging, treatment planning, and management of cancer.

$^{18}$F labeled molecules, including peptides and agents for tracking gene therapy, have resulted in several new radiopharmaceuticals. Other short-lived PET radionuclides, mainly $^{11}$C and to a lesser extent $^{15}$O, are also being studied, despite the logistical problems due to their short half-lives. Other radionuclides used in diagnosis are: $^{133}$Xe, $^{51}$Cr, $^{75}$Se, $^{123}$I, $^{125}$I, $^{59}$Fe, $^{57}$Co, and $^{58}$Co. Radionuclides common used in diagnostic are shown on the right in Table 3.2.

In vitro studies procedures are performed in test tubes in laboratory and allow more flexibility in the energy range. Radioimmunoassay (RIA) is a special type of in vitro procedure that combines the use of radiochemical and antibodies to measure the levels of hormones, vitamins, and drugs in a patient's blood. Radioimmunoassay has many applications in blood banking, diagnosis of allergies and endocrinology.

In vitro studies use three important pure gamma emitters $^{125}$I, $^{57}$Co, and $^{51}$Cr for hematology and endocrinology tests, and pure beta emitters $^{32}$P, $^{3}$H, $^{35}$S, $^{14}$C, and $^{45}$Ca for radioimmunoassay techniques.

Therapeutic procedures in nuclear medicine employ beta emitter radionuclides—which produce a high ionization in a very short path length—though they could release a high dose in areas where radiotracer acquiring is intracellular and highly selective. $^{131}$I continues being used for cost effective treatment of hyperthyroidism and metastatic thyroid cancer. Some bone seeking radiopharmaceuticals, such as $^{32}$P (as sodium phosphate) (FDA approved), $^{89}SrCl_2$, $^{153}$Sm-EDTMP (FDA approved) and $^{186}$Re-HEDP (Europe approved) are increasingly used as cost effective bone pain palliative agents of osteoblastic metastases. $^{131}$I-mIBG and

**Table 3.2** Examples of radionuclides used in nuclear medicine

| Clinically used radionuclides in cancer therapy | | | Common used radionuclides in diagnostic | | |
|---|---|---|---|---|---|
| Radionuclide | Pharmaceutical | Clinical use | Radionuclide | Half-life | Examination |
| $^{131}I$ | NaI | Differentiated thyroid carcinomas | $^{99m}Tc$ | 6 h | Heart studies |
| $^{32}P$ | $NaH_2PO_4$ | Polycythemia vera | $^{123}I$ | 13 h | Thyroid studies |
| $^{89}Sr$ | $SrCl_2$ | Bone metastases | $^{201}Tl$ | 78 h | Myocardial studies |
| $^{131}I$ | mIBG | Neural crest tumors | $^{11}C$ | 20 min | Brain imaging |
| $^{153}Sm$ | EDTMP | Bone metastases | $^{111}In$ | 67 h | Brain studies |
| $^{186}Re$ | HEDP | Bone metastases | $^{67}Ga$ | 78 h | Tumor studies |
| $^{32}P$ | $CrPO_4$ | Intracavitary | $^{81m}Kr$ | 13 s | Lung studies |
| $^{90}Y$ | Microspheres | Hepatic tumors | $^{13}N$ | 10 min | Heart studies |
| $^{90}Y$ | Antibodies | Various tumors | $^{15}O$ | 2 min | Oxygen studies |
| $^{114m}In$ | Lymphocytes | Lymphoma | $^{18}F$ | 110 min | Epilepsy |
| $^{131}I$ | Antibodies | Various tumors | | | |
| $^{131}I$ | Lipiodol | Hepatic tumors | | | |

$^{131}I/^{188}Re$ labeled lipiodol continues to attract interest for treatment of neuro-endocrine tumors and hepatocellular carcinoma, respectively. Radionuclides common used in cancer treatments are shown on the left in Table 3.2.

Radionuclides have been used for biomedical research in studies of coronary artery diseases; microbiological studies of infectious diseases with labeled leukocytes; tumor studies with labeled molecules, monoclonal antibodies and other receptor-avid molecules to more specifically target tumors in oncology; radioimmunotherapy (therapeutic agents based on labeled monoclonal antibodies); diagnostic studies of cerebrovascular diseases; diagnosis and treatment of thyroid cancer and other endocrine and blood disorders; bone malignancy therapy; reduction of pain in bone diseases; treatment of malignant effusions (in both pleural and peritoneal spaces); as adjuvant therapy in patients with ovarian carcinoma; in the studies of renal functioning; and in planning and monitoring of stereotactic surgery treatments; etc.

The labeling of biochemical and neurochemical molecules of interest, combined with the use of positron emission tomography (PET) technique, have allowed a more accurate specification of the brain functional organization at the level of massive neuron clustering, its networks and systems.

The worldwide spread of medical applications of radiation sources in the last century has been remarkable. Just mentioning some numbers, all over the world there are more than 10,000 linear accelerators for radiation therapy, annually are done about 3.6 billion of *X*-ray examinations; near 33 million of diagnostic nuclear medicine procedures in vivo, and about 5.1 million of radiotherapy treatments [1]. From 20 to 25 % approximately of all PET medical institutions in the USA and Western Europe have at least one baby cyclotron to take advantage of the full spectrum of available PET isotopes (2 h to few seconds) [2].

## 3.2 Radiation Sources in Industry and Agriculture

Radiation sources have contributed to the solution of a wide variety of problems in industry and agriculture. They are associated with instrumentation and process control; measurement technology; multielemental analysis and characterization of materials; stimulation of plant growth; food preservation; plague control; plant mutation; industrial radiography; well logging; vulcanization of polymers; resin radio polymerization; water purification; and so forth. Table 3.3 shows some of the radioactive sealed sources that are commonly used in industrial and agricultural applications.

Beneficial use of radiation sources is primarily based on the easiness and reliability of its detection; on specifics of interaction of radiation with different materials; and on the activation which can be induced by radiation.

Industrial radiography is one of the most widespread applications. It takes advantage of the characteristic of certain radiation to penetrate materials in depth to view them in a way that cannot be seen otherwise. Industrial radiography is a key component for nondestructive testing; inspection of weld and weld overlays in piping and reservoirs (oleoducts, gas pipelines, petrochemical plants, thermoelectrical power stations, nuclear power stations, etc.); for determination of stress fields in structural components, as in bridge building; and for wear and corrosion monitoring and control, etc. Depending on the material thickness and the purpose of examination the sources used are *X*-ray generators or gamma sources.

Neutron sources have been used, e.g., for moisture content measurement and location of reinforced bars in concrete inspection. Using Betatrons and linear accelerators to produce combined *X*-ray photons of high energy, in conjunction with computational techniques, have allowed completion of *X*-rays images in real time, very useful for the analysis of components for aeronautic and nuclear industry.

Geotechnical investigations, as well as exploration and exploitation of mineral resources and hydrocarbons, have required the use of gamma and neutron sources to gain information that cannot be obtained in any other way. Main measurements are density, porosity, and moisture, as well as hydrocarbon content. Neutron sources commonly used are $^{241}$Am-Be with maximum activities of 800 GBq,

**Table 3.3** Sealed sources in common industrial and agricultural applications

| Application | Radionuclide | T½ | Source activity |
|---|---|---|---|
| Industrial radiography | $^{192}$Ir | 74 d | 0.1–5 TBq |
| | $^{60}$Co | 5.3 y | |
| | $^{137}$Cs | 30 y | |
| | $^{170}$Tm | 128.6 d | |
| | $^{169}$Yb | 32 d | |
| Well logging (detailed record or diagraphy) | $^{241}$Am–Be | 433 y | 1–800 GBq |
| | $^{238}$Pu–Be | 93 y | 100–200 GBq |
| | $^{252}$Cf | 2.6 y | $10^7$–$10^9$ n.s$^{-1}$ |
| | $^{137}$Cs | 30 y | 37 MBq–100 GBq |
| Moisture measurement | $^{241}$Am–Be | 433 y | 0.1–2 GBq |
| | $^{252}$Cf | 2.6 y | |
| | $^{226}$Ra–Be | 1,600 y | |
| Density measurement | $^{137}$Cs | 30 y | 0.1–100 GBq |
| | $^{60}$Co | 5.3 y | 1–100 GBq |
| | $^{241}$Am | 433 y | 1–10 GBq |
| Level measurement | $^{137}$Cs | 30 y | 0.1–20 GBq |
| | $^{60}$Co | 5.3 y | 0.1–10 GBq |
| Thickness measurement | $^{90}$Sr | 29 y | 0.1–4 GBq |
| | $^{85}$Kr | 10.8 y | 0.1–50 GBq |
| | $^{14}$C | 5,730 y | |
| | $^{147}$Pm | 2.6 y | |
| | $^{241}$Am | 433 y | |
| Radioactive lightning rods or arresters | $^{241}$Am | 433 y | 50–500 MBq |
| | $^{226}$Ra | 1600 y | |
| Static charge eliminators | $^{241}$Am | 433 y | 1–4 GBq |
| | $^{210}$Po | 138 d | 1–4 GBq |
| Smoke detectors | $^{241}$Am | 433 y | 0.02–3 MBq |
| | $^{85}$Kr | 10.7 a | |
| | $^{239}$Pu | 2.4 × $10^4$ y | |
| *X*-ray fluorescence | $^{55}$Fe | 2.6 y | 0.1–5 GBq |
| | $^{109}$Cd | 463 d | 1–8 GBq |
| | $^{238}$Pu | 93 y | |
| | $^{241}$Am | 433 y | |
| | $^{57}$Co | 270.9 d | |

(continued)

**Table 3.3** (continued)

| Application | Radionuclide | T½ | Source activity |
|---|---|---|---|
| Food preservation, sterile insect technique, product radio mutagenesis, and radio sterilization | $^{60}$Co | 5.3 y | 0.1–400 PBq |
| | $^{137}$Cs | 30 y | 0.1–400 PBq |
| Wear control of blast furnaces | $^{60}$Co | 5.3 y | 2 GBq |
| Calibration facilities | $^{60}$Co | 5.3 y | 1–100 GBq |
| | $^{137}$Cs | 30 y | 1–100 GBq |
| Mass measurement systems on conveyor belts | $^{137}$Cs | 30 y | 0.1–40 GBq |

$^{238}$Pu-Be, $^{239}$Pu-Be, and $^{226}$Ra-Be. $^{137}$Cs gamma sources are used in conjunction to measure the density of the formation.

Nucleonic gauges or nucleonic control systems is another area of industrial application of ionizing radiation in which significant benefits have been reported over the last 50 years [3]. Just mentioning some examples, radiation sources are used to control the level of liquids, solids, slurries, and pulps; the size and density of materials and catalyst beds; for moisture measurements; for porosity control; pollution analysis and measurement of environmental parameters; overlay control in semiconductor manufacturing; and to measure thickness of coatings, metal components, crusts, etc.

Gamma Computed Tomography (CT) is a fast developing technique focused on imaging, capable of measuring multicomponent and multiphase processes carried out in a wide range of industries. Gamma tomography allows measuring spatial distributions of material based on its attenuation properties [2, 4]. It has been used to measure the spatial density distribution inside processing vessels or pipelines.

*X*-ray diffraction and *X*-ray fluorescence are commonly used for phase and structural analysis of metals, minerals, and other materials in industry. Activation induced by neutrons—neutron activation analysis (NAA)—is one of the most sensitive and accurate methods of analysis for meeting industrial requirements of trace elements. Neutron resonance radiography (NRR) and neutron radiography are also powerful tools for nondestructive testing of materials [5]. Positron annihilation spectrometry is very useful in nanotechnology for identification of crystal punctual defects.

Commercial facilities in chemical industry use high intensity gamma sources of various tens of PBq for radiation induced polymerization and crosslinking, and tire component curing. Gamma sources have also been used for environmental protection applications as flue gas, water and wastewater treatment, as well as sewage sludge hygienization. Food industry uses $^{60}$Co radiation sources to preserve food and eliminate food-borne pathogens [6, 7].

Electron beams (maximum energy of 10 MeV) and *X*-ray irradiation systems based on accelerator technology have also become a real possibility in the last decades. Worldwide, there are over 1,400 high-current industrial electron beam accelerators in commercial use [8]. In the United States, the Postal Service is processing mail with electron beam technology to eliminate anthrax.

Portable neutron and gamma gauges are also used to control soil moisture and density for irrigation. High activity $^{60}$Co and $^{137}$Cs gamma sources are typically employed for radio stimulation of plant growth and seed germination, conservation of harvested products, and induction of sterility of males for pest eradication and control (sterile insect technique). Radiation induced mutations—radiation breeding—has also produced thousands of useful mutants and a sizable fraction of the world's crops. The mutations can improve yield, quality, taste, size, and resistance to disease and can help plants adapt to diverse climates and conditions.

Radionuclide tracer techniques have been widely used throughout the industry in important studies to optimize processes, solve problems, and improve product quality, e.g., determining filtering efficiency and flow rate of filtration; leak detection; dynamic flow rate measurements; determining mixing efficiency, and homogeneity of blend operations; etc. Some radionuclides used in tracer techniques have been $^{24}$Na, $^{46}$Sc, $^{51}$Cr, $^{64}$Cu, $^{82}$Br, and $^{35}$S. Radionuclide generators designed specifically for industrial purposes are now being evaluated to improve the supply of short-lived radionuclides with which to carry out extended studies.

Radiotracer techniques have been used in petroleum reservoirs to understand the fluid dynamic regime on oilfields; as well as for leak detection in underground pipelines [9]. Radionuclides used in petroleum tracer techniques include $^{14}$C, $^{3}$H, $^{32}$P, $^{35}$S, $^{133}$Xe, $^{99m}$Tc, and $^{85}$Kr.

Radionuclide tracer techniques are also useful in agriculture, hydrology, and hydrography. Radiotracers have helped to learn about fertilizer distribution within soil. They have also allowed identify the location of underground watersheds and its derivations, and to better understand sedimentation, erosion, and desertification occurrences. Nucleotides labeled with $^{33}$P and $^{32}$P, as well as amino acids and proteins labeled with $^{131}$I have been used in studies for improving livestock productivity. Most important radionuclides used on these applications are $^{32}$P, $^{15}$N, $^{14}$C, and $^{3}$H.

## 3.3 Radiation Sources in Research and Education

Most of the radiation sources used in research was mentioned above in Table 3.3. Some other sources commonly found in research and academic laboratories are listed in Table 3.4. They include sources for testing and calibrating measuring instruments like radiometers, dosimeters, and spectrometers, as well as standard and control sources. Calibration sources vary from some tens or hundreds of kBq to some tens of GBq.

Mössbauer spectrometry is a versatile technique capable of give information about chemical, structural, magnetic, and time-dependent properties in both crystallographic and amorphous materials. It is widely used in a large number of research centers and universities.

*X*-ray diffraction analysis is a nondestructive technique useful to get information about the crystallographic structure, chemical composition, and various physical

**Table 3.4** Some sealed sources commonly used in laboratories

| Application | Radionuclide | T½ | Activity |
|---|---|---|---|
| Test and calibration | Various | | <0.1 GBq |
| Irradiation | $^{60}$Co | 5.3 y | 1–100 TBq |
| | $^{137}$Cs | 30 y | |
| Calibration benches | $^{137}$Cs | 30 y | <100 TBq |
| | $^{60}$Co | 5.3 y | <100 TBq |
| | $^{252}$Cf | 2.6 y | <10 GBq |
| | $^{241}$Am–Be | 433 y | |
| | $^{238}$Pu–Be | 93 y | |
| Mössbauer spectrometry | $^{57}$Co | 272 d | 37–3700 MBq |
| Electron capture detectors | $^{63}$Ni | 100 y | 200–500 MBq |
| | $^{3}$H | 12.3 y | 1–50 GBq |
| Tritium targets | $^{3}$H | 12.3 y | 1–10 TBq |

properties of materials and thin films. This technique is very advantageous for analysis of metals, alloys, and minerals. *X*-ray fluorescence is more helpful for elemental analysis and trace chemical analysis. By *X*-ray fluorescence traces of some elements like P, S, Sc, V, Cr, Mn, Co, Ni, Zn, Rb, Sr, Y, Zr, Ba, and Pb may be identified in biological samples, water, minerals, and oil. It is particularly effective for research in geochemistry, forensic science, and archeology.

Neutron generators and isotopic neutron sources like $^{241}$Am-Be, $^{239}$Pu-Be, and $^{226}$Ra-Be also have application in research and educational centers. Neutron activation analysis is a highly efficient technique mostly used for prospection and examination of precious metals, silicates, and clays; for pollutant detection; and for analysis of complex biological samples. Most nuclear reactions used for neutron generation require high-energy particles and produce fast neutrons, i.e., neutrons with energies of several MeV. Neutron generators targets are thin films of metal in form of stable chemical compounds called metal hydrides, made up of two hydrogen (deuterium or tritium) atoms per metal atom.

Large research and academic institutions might have other important sources like nuclear research reactors, particle accelerators, cyclotrons, and other irradiation facilities. High activity $^{137}$Cs and $^{60}$Co gamma sources (hundreds of GBq to hundreds of TBq) are used in researches in radiobiology and radiation chemistry; electron beams are used in medical, biological, physical and environmental studies.

It is possible to use the decay properties of natural radionuclides to determine the relative age of underground waters, ices, rocks, sediments, and other natural occurrences. Cosmogenic ($^{18}$O, $^{13}$C, $^{2}$H) and anthropogenic ($^{3}$H, $^{137}$Cs) radionuclides are used in climatology studies to establish temperature and humidity patterns, ocean tides, atmosphere and ocean dynamic. It has been possible to establish a detailed chronology of ocean temperature changes during the main part of Quaternary and to track huge mass of water when using, for example, $^{14}$C dating, uranium-series disequilibrium dating, and potassium-argon dating. Natural occurred

radionuclides, as those from the decay chain of radium and uranium, have been used for dating sediments from deep seas and from lakes, and for dating cave deposits.

Radiotracer techniques have also made a huge contribution to medical, chemical, and biological research. The high measuring sensitivity and the labeling selectivity of tracer techniques, along with the fact that the tracer does not affect the properties or behavior of the sample, have made possible the labeling of biological molecules and the observation of living system in real time. Radioactivity contribution to life science must be qualified as tremendously positive. Studies with tracer techniques can be conducted today at all levels of biological organization—molecular, cellular, multicellular (tissues), organs, and organ system. A new era in molecular nuclear medicine is emerging with the support of molecular imaging, imaging of gene expression, and receptor-based radiopharmaceuticals [10]. It is an opening to the study of biochemical processes of proteins as they carry out instructions from genetic coding.

## 3.4 Risks and Its Perception

Risk may be used to express a quantity—a probability—or a quality indicating the possibility of harm. In radiation safety risk is a "multiattribute quantity expressing hazard, danger or chance of harmful consequences, associated with actual or potential exposures. It relates to quantities such as the probability that specific deleterious consequences may arise and the magnitude and character of such consequences [11]." ["Risk definition is reprinted from GSR Part 3 Radiation Protection and Safety of Radiation Sources: International Basic Safety Standards, by IAEA, 2014, with IAEA permission."] In other words, for radiation safety purposes, risk is the probability of detrimental health effects in a person or group, or their progeny, as a result of exposure to radiation, including the likelihood of the occurrence of such effects. This definition has two concepts involved; one is the likelihood of an event to occur; the second is the probability of health effect or consequence to appear. Since risk involves a full range of likelihoods and consequences which are not certain to occur, the risk acceptability will depend mostly on the benefit associated to the event or circumstance the risk has arisen from.

Like any other human activity, exposure to ionizing radiation involves certain level of risk. The level of risk is comparable to and even lower than the level of risk each individual is assumed to accept for enjoying the benefits of modern lifestyle. It is a fact that the use of radiation sources, along with nuclear power generation, has been beneficial to the humanity and its further development. Actually, the discovery of radioactivity and nuclear energy was one of the greatest milestones of the 20th century.

Unfortunately, radiation risk has not been always well accepted. Perhaps the initial belief on the beneficial effect of *X*-rays and radium was soon reverted into apprehension and fear to nuclear power after the war [12]. The fact reflects a mix of

dissimilar emotional and affective responses, cognitive factors, and confidence issues mostly confusedly associated with military use of nuclear energy. There are, however, some objective reasons that might have contributed to reinforce this perception of radiation risk. First, the lack of positive information about radiation which produces a dubious credibility in the public on authority and experts' judgments and decisions in case of accident, in addition to our own inability of an effective communication. Second, the complex social situations that an accident creates in the population as a result of its life disruption, and public's uncertainties around whether or not the risk could be controlled.

An effective radiation safety program[5] at any facility can assure no harm to workers, public, and environment during its normal operation—meaning by facility a nuclear power plant, any stage of nuclear fuel cycle, an irradiation unit at a plant, a radiotherapy division at a hospital, etc. Doses received by workers are far below from those leading to tissue reactions and are near to background radiation, though no effect may be expected. The following data will help to understand the dose levels when comparing each other.

| | |
|---|---|
| Average dose to individuals from terrestrial sources of radiation | 330 μSv/year |
| Average dose to individuals from potassium inside the human body | 300 μSv/year |
| Average background dose to individuals from all natural sources | 2.4 mSv/year |
| Dose from an intercontinental fly (between Europe and North America) | 50 μSv |
| Average dose from *X*-rays medical uses | 400 μSv/year |
| Average dose from living within 8 miles from a nuclear power station | 0.2 μSv/year |

Similar data are illustrated in Fig. 3.2 [13]. Occupational exposure contribution to the average dose of the world population is negligible compared to the background dose from all natural sources. The average dose from diagnostic radiology in the period 1997–2007 was 0.62 mSv per caput (an average dose of 1.92 mSv for countries with healthcare systems at level I). For the United States population, the annual per caput effective dose increased from 3.0 mSv in 1980 to 6.2 mSv in 2006 due to CT scanning, making medical exposure comparable to that from natural background [1].

The average measurable dose (the total effective dose divided by the number of individuals receiving a measurable dose) in the U.S.A per worker for commercial nuclear reactors and other facilities calculated from reported data, decreased from 2 mSv in 2006 to 1.7 mSv in 2010 [14].

If considering the approximated overall fatal risk coefficient recommended by the International Commission on Radiological Protection (ICRP) of 5 % per Sv [15] for levels of doses which are a fraction of the natural background radiation dose, the estimated probability for cancer risk would be of the order of $\sim 10^{-5}$ %. In

[5]An effective radiation safety program implies practical implementation of all principles and requirements stated by national and international standards.

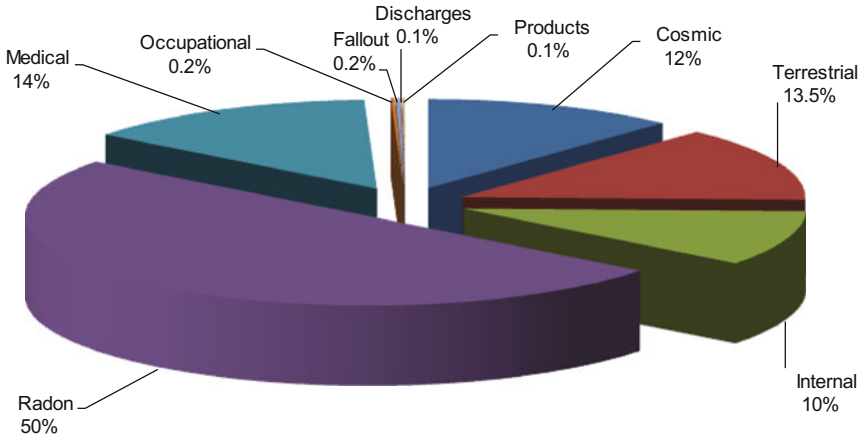



**Fig. 3.2** Population exposure from all sources of radiation in UK 2003

2012, just for comparison, the worldwide average of cancer deaths was $\sim 1.04 \times 10^{-3}$.

But the facts "that ionizing radiation is commonly poorly understood by the general public, that it can cause harm without being seen or felt, is capable of causing cancer and is associated in the public mind with atomic explosions and uncontrolled accidents, has, as a consequence, that public aversion to technologies involving radiation tends to be particularly strong, and this has been reinforced in some cases by poor communication and by incorrect and biased information [16]." ["Reprinted from IAEA INSAG 11 A, The Safe Management of Sources of Radiation: Principles and Strategies 1999 with IAEA permission."] Exceptions are diagnostic and therapeutic uses of radiation where individuals benefit themselves from exposure.

Although they are rare and radiation sources are well controlled in a normal operation, accidents may occur. Depending on the source, its activity and the circumstances of the accident, individuals could receive doses within the limits established for workers and public, might be tolerated and is expected that will not cause any harm. In extraordinary circumstances like in emergency situations, doses to individuals could be higher, reach some values within the range of tissue reactions (deterministic effects) and health effects may be expected. For example, among the 600 workers present on the site the day of the Chernobyl accident—134 of them confirmedly diagnosed with acute radiation syndrome (ARS)—28 persons died in 1986 due to ARS, and 19 more persons died in the period 1987–2004 from different causes [17, 18].

To evaluate the risk that could be imposed on individuals and on members of the public if the control of radiation source is lost, it is important to consider specific criteria as the size, type, and activity of the source; the complexity involved in the application (medical, industrial, power generation, radioactive waste management,

etc.); the capacity for regaining an effective control; and the overall potential for harm. Potential for harm involves the number of individuals who might be affected by dose or release of radioactive material, including tends and pathways for release, magnitude of any individual dose and nature of the exposure (internal or external), the spread and form of potential contamination, and resulting economic and social consequences.

Unsealed sources in laboratories, or for tracer techniques, are of low activity, and/or sufficient diluted, or contained in a limited small volume so that, in case of a spill, the expected consequence—contamination—is usually limited to a specific room or area. People may get contaminated, but harmful health effects are usually not to be expected. People must be evaluated if incorporation through any known pathway has occurred.

Sealed sources used in, e.g., radiotherapy or industrial radiology, have high specific activities and deliver high doses. If any of these sources happen to be out of their housing and shielding, health effects to individuals near or in the vicinity of the source could be expected. If the source enclosed capsule is damaged or chemically altered and its contents dispersed, contamination of areas, existing objects and/or individuals may be anticipated. Harmful health effects are to be considered and evaluated.

Risk control is aimed to protect individuals, society as a whole, and the environment, not to scare of ionizing radiation. Still, the general public, and even some people who work with radiation, admit an irrational fear mainly based on the consequences of potential accidents. This explains why any news regarding an incident with radioactive materials, trivial or not, may appear as a disaster when clear and accurate information is not provided.

Some concepts are frequently misunderstood and, even worse, wrongly communicated. For example, radiation exposure at nearly normal existing levels is often confused with irradiation to an unacceptable or harmful level of radiation. Most people do not distinguish penetrating radiation (*X* and gamma rays, neutrons) from particle radiation, and it is often heard that "a contamination occurred" when the shielding of a sealed source had been damaged or a nuclear reactor had had any operational issue. Furthermore, some people believe that you could be "contaminated" by *X*-rays, that irradiated food is poisoned, and that exposure causes impotence in males.

Public reaction to the risk of ionizing radiation is different depending on the application. When a hospital or a clinic uses a radiodiagnostic source, most people do not think of the risk, they think of the benefit. It happens that many patients ask to be referred to a CT scan instead of an alternative diagnostic procedure. Quite the opposite, when using a radiation source, e.g., for nondestructive testing, weld inspection, nuclear instrumentation, or process control, it is frequently regarded as something "dangerous," or something best not to deal with. Public risk perception concerning nuclear energy is particularly averse, despite that nuclear power plants have accumulated more than 8,700 reactor-years of satisfactory operation.

Surveys are an excellent way to assess risk perception. Some studies have focused on this topic in the last decades and interesting results have been obtained.

A nuclear reactor, or radioactive waste facility accident, has been considered by the public as a dreadful, unfamiliar, catastrophic, and uncontrollable risk; while saccharin consumption, taking aspirin, antibiotics and oral contraceptives, and even alcohol consumption, have been paradoxically considered by many as voluntary and controllable risks, with negligible consequences to future generations [19].

Statistical studies had confirmed that general public considers radiation exposure risk above other well-known risks as smoking, car accidents, and armory carry. They had also confirmed that the public does not recognize nuclear energy benefits, but accept *X*-rays benefits. Public perception of health risk derived from indoor radon inhalation is likewise lesser than what is being demonstrated today [20].

These facts emphasize the relevance of ample information on the benefits of ionizing radiation and reinforcing the general public awareness of and involvement in safety culture matters. Differences between exposure due to normal operation, potential exposures, and emergency and existing exposure situations are important, as well as the level of doses related to them.

## References

1. UNSCEAR, 2008 (2010) Sources and Effects of Ionizing Radiation. Report to the general assembly with scientific annexes, vol I, Scientific annex a medical radiation exposures. UN, New York
2. International Atomic Energy (2006) Nuclear technology review, IAEA/NTR/2006. IAEA, Vienna
3. International Atomic Energy (2005) Technical data on nucleonic gauges, IAEA-TECDOC-1459. IAEA, Vienna
4. International Atomic Energy (2008) Industrial process gamma tomography, Final report of a coordinated research project 2003–2007, IAEA-TECDOC-1589. IAEA, Vienna
5. International Atomic Energy (2008) Neutron imaging: a non-destructive tool for materials testing. Report of a coordinated research project 2003–2006 IAEA-TECDOC-1604. IAEA, Vienna
6. International Atomic Energy (2004) Emerging applications of radiation processing. In: Proceedings of a technical meeting held in Vienna, 28–30 April 2003. IAEA-TECDOC-1386. IAEA, Vienna
7. International Atomic Energy (2005) Gamma irradiators for radiation processing. IAEA, Vienna
8. International Atomic Energy (2011) International irradiation association. Industrial radiation processing with electron beams and X-rays. IAEA, Vienna2011
9. International Atomic Energy (2004) Radiotracer applications in industry—a guidebook, Technical report series no. 423. IAEA, Vienna
10. Kim EE (2004) Book review—molecular nuclear medicine: the challenge of genomics and proteomics to clinical practice. J Nucl Med 45(8):1422
11. International Atomic Energy (2014) Radiation protection and safety of radiation sources: international basic safety standards, iaea safety standards series GSR part 3. IAEA, Vienna
12. Kellerer AM (2002) Invisible threat. The risk of ionizing radiation, Chapter for NUPECC
13. Department for Environment, Food and Rural Affairs (DEFRA) U.K. (2003) e-digest statistics about: radioactivity. http://archive.defra.gov.uk/evidence/statistics/environment/radioact/
14. NRC NUREG-0713 (2010) Annual report on occupational radiation exposure, vol. 28, 2007 and vol. 32, 2010

15. ICRP (2007) The 2007 recommendations of the international commission on radiological protection. Annals ICRP 37(2–4)
16. International Atomic Energy (1999) The safe management of sources of radiation: principles and strategies, INSAG 11 A. IAEA, Vienna
17. WHO (2006) Health effects of the chernobyl accident and special health care programmes, Report of the UN Chernobyl forum expert group "health". WHO, Geneva
18. UNSCEAR (2011) Sources and effects of ionizing radiation—UNSCEAR 2008 report. Volume II annex D—health effects due to radiation from the Chernobyl accident. United Nations, New York
19. Proceedings of a workshop, Oslo, Norway (1996). Radiation risk. Risk perception and social constructions. Radiat Prot Dosim 68(3/4):165–180
20. Proceeding of a conference London (1996) Becquerel's legacy: a century of radioactivity. Radiat Prot Dosim 68(1/2):37–42

# Chapter 4
# Basic Quantities and Units in Radiation Safety

Human senses are not capable of detecting ionizing radiation, thus its detection has to be done indirectly by the effects it causes. Ionizing radiation is the energy that comes from a source, travels through a medium and may be absorbed by it. Therefore, to characterize and measure ionizing radiation, we need quantities that describe the source, the radiation field at the point of interest, and the energy deposited in the material with which ionizing radiation interacts. Selection of the most appropriate quantity depends on the specific case. The values assigned to the various quantities may be obtained by calculations and/or measurements.

The International Commission on Radiological Protection (ICRP) and the International Commission on Radiation Units and Measurements (ICRU) are international organizations who develop internationally accepted recommendations on radiation measurements, quantities, and units. The ICRU defines the units, and the ICRP recommends how they are used for radiation protection. In the United States, the main counterpart is the National Council on Radiation Protection and Measurements (NCRP), an organization chartered by the U.S. Congress.

## 4.1 Source Quantities

The starting point to evaluate the amount of radiation from a specific exposure situation is the source. A radioactive source—a radioactive material used as a source of radiation—is characterized by its activity and half-life.

Activity is "the quantity $A$ for an amount of radionuclide in a given energy state at a given time," ["Reproduced with permission by the IAEA."] also defined as the rate at which nuclear transformations occur [1]:

$$A(t) = \frac{\mathrm{d}N}{\mathrm{d}t}, \quad \text{unit: } \mathrm{s}^{-1}$$



H. Domenech, *Radiation Safety*, DOI 10.1007/978-3-319-42671-6_4

In other words, activity is the number of disintegrations of a radionuclide per unit time.

Where d$N$ is the expectation value of the number of spontaneous nuclear transformations from the given energy state in the time interval d$t$. The International System of Units (hereafter referred to as SI) unit of activity is the reciprocal second ($s^{-1}$), having the special name *Becquerel* (Bq): 1 Bq = 1 $s^{-1}$

Becquerel replaces formerly activity expressed in *curies* (Ci), where 1 Ci = 3.7 $\times$ $10^{10}$ Bq. Activity values may be given in Ci (with the equivalent in Bq in parentheses) if they are being quoted from a reference that uses Ci as the unit.

For each radionuclide, the physical quantity half-life ($T_{½}$) is the time required for the nuclei population to decrease by a radioactive decay process by half. The activity also decreases by half by the same radioactive decay process, so the time taken for activity to decrease by half can be used as an alternative definition of half-life. Where it is necessary to distinguish this from other half-lives (e.g., biological half-life, effective half-life), *radioactive half-life* is used. The SI unit is the time unit (seconds, minutes, hours, or years).

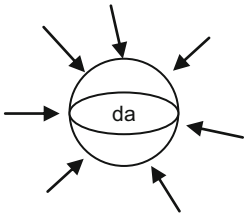
$$T_{1/2} = \frac{\ln 2}{\lambda},$$

where $\lambda$ is the decay constant or disintegration constant for a radionuclide in a particular energy state, in $s^{-1}$.

Quantities used to describe a radiation field from either a radioactive source or a monoenergetic radiation beam produced by a machine (e.g., a photon beam, an electron beam, a neutron beam, etc.), are fluence as a measure of the density of particles in the radiation field, energy fluence as a measure of energy transported by particles in the field, particle fluence rate, and energy fluence rate.

Fluence—or particle fluence—at a given point in space is the number of particles d$N$ incident in a given time on a small sphere of cross-sectional area d$a$ centered at that point, divided by the cross-sectional area of that sphere:

$$\Phi = \frac{\mathrm{d}N}{\mathrm{d}a}, \quad \text{unit: } \mathrm{m}^{-2}$$




As shown in the illustration, a sphere of cross-sectional area expresses the fact that the area is perpendicular to the direction of each particle and hence that particle fluence does not depend on the incident angle of the radiation. In contrast, and it is

important to take into account, planar particle fluence is the number of particles crossing a plane per unit area and hence depends on the angle of incidence of the particle beam.

Energy fluence is the quotient of d$E$ by d$a$, where d$E$ is the radiant energy incident on a sphere of cross-sectional d$a$. Energy fluence can be calculated from particle fluence using the following expression:

$$\Psi = \frac{\mathrm{d}N}{\mathrm{d}a} \cdot E = \Phi \cdot E, \quad \text{unit: J.m}^{-2},$$

where $E$ is the energy of the particle and d$N$ represents the number of particles with energy $E$.

Realistic photon or particle beams are almost all polyenergetic and energy distributions are frequently required. The energy distribution of fluence and energy fluence in the interval between $E$ and $E$ + d$E$ is given by

$$\Phi_E(E) = \frac{\mathrm{d}\Phi}{\mathrm{dE}}(\mathrm{E}), \quad \text{unit: m}^{-2}.\text{J}^{-1}$$

and

$$\Psi_E(E) = \frac{\mathrm{d}\Psi}{\mathrm{dE}}(\mathrm{E}) = \frac{\mathrm{d}\Phi}{\mathrm{d}E}(E) \cdot E, \quad \text{unit: m}^{-2}$$

Fluence rate (or flux density) is the quotient of particle fluence d$\Phi$ by d$t$, where d$\Phi$ is the increment of the fluence in time interval d$t$:

$$\phi = \frac{\mathrm{d}\Phi}{\mathrm{d}t}, \quad \text{unit: m}^{-2}.\text{s}^{-1}$$

Likewise, the energy fluence rate (also referred to as intensity) is the quotient of d$\Psi$ by d$t$, where d$\Psi$ is the increment of the energy fluence in the time interval d$t$:

$$\Psi = \frac{d\Psi}{\mathrm{d}t}, \quad \text{unit: J.m}^{-2}.\text{s}^{-1}$$

## 4.2 Interaction Coefficients and Related Quantities

The Linear Energy Transfer (LET)—or restricted linear collision stopping power—describes the interaction of a radiation field of charged particles of a given type and energy with a material. Linear energy transfer is the quotient of d$E_\Delta$ by dl:

$$L_\Delta = \frac{\mathrm{d}E_\Delta}{\mathrm{d}l}, \quad \text{unit: J.m}^{-1},$$

where $\mathrm{d}E_\Delta$ is the mean energy lost by the charged particles due to electronic interactions in traversing a distance dl minus the mean sum of the kinetic energies in excess of $\Delta$ of all the electrons released by the charged particles.

In other words, LET is the radiation energy lost per unit length of path through a material. A high value of linear energy transfer indicates that energy is deposited within a small distance. LET is an important factor to consider when evaluating the absorbed dose from an exposure, but is rarely measured and must be calculated from theory.

For indirectly ionizing radiation such as photons, a linear energy-transfer coefficient, $\mu_\Delta$, is used to describe the energy transferred into kinetic energy of secondary charged particles released by expecting interactions in traversing a distance $\mathrm{d}l$ in the material, thus

$$\mu_\Delta = \frac{\mathrm{d}E\Delta}{N.E.\mathrm{d}l}, \quad \text{unit: m}^{-1},$$

where $E$ is the energy of uncharged particles, $N$ is the number of uncharged particles incident on a material of length $\mathrm{d}l$, and $\mathrm{d}E_\Delta$ is the sum of kinetic energies of all charged particles released in the length $\mathrm{d}l$ by uncharged particles. $\Delta$ is an energy transfer threshold. Linear energy-transfer coefficient represents the probability per unit length that energy is transferred to charged particles.

The mass energy-transfer coefficient, $\mu_{\Delta m}$, for uncharged particles, is the quotient of the linear energy-transfer coefficient by the density $\rho$ of the absorbing material $m$. It depends on material, and the type and energy of radiation. The mass energy-transfer coefficient when multiplied by the photon energy fluence gives the dosimetric quantity kerma.

$$\mu_{\Delta m} = \frac{\mu_\Delta}{\rho}, \quad \text{unit: m}^2.\text{kg}^{-1}$$

The mass attenuation coefficient, $\mu/\rho$, and the mass energy-absorption coefficient, $\mu_{ab}/\rho$, are basic quantities used in calculations of the penetration and the energy deposition by photons ($X$-ray, gamma rays, bremsstrahlung) in biological, shielding, and other materials for many scientific, engineering, and medical applications.

The mass attenuation coefficient of a material for uncharged particles is the quotient of $\mathrm{d}N/N$ by $\rho\ \mathrm{d}l$, where $\mathrm{d}N/N$ is the fraction of particles that experience interactions in traversing a distance $\mathrm{d}l$ in the material of density $\rho$, thus

$$\frac{\mu}{\rho} = \frac{1}{\rho \mathrm{d}l}\frac{\mathrm{d}N}{N}, \quad \text{unit: m}^2.\text{kg}^{-1}$$

$\mu$ represents the sum of attenuations coefficients for all individual interactions that a photon may have with atoms of the absorber—Rayleigh and Compton scattering, photoelectric effect, pair production, and photonuclear reaction.

Tables and graphs of the mass attenuation coefficients for all of the elements $Z$ = 1–92, and for compounds and mixtures of radiological interest, are available at the Physics Laboratory of the National Institute of Standards and Technology website [2].

The mass energy-transfer coefficient, $\mu_{\Delta m}/\rho$ and mass energy-absorption coefficient, $(\mu_{ab}/\rho)$ are related through the following relationship:

$$\frac{\mu_{ab}}{\rho} = \frac{\mu_{\Delta m}}{\rho}(1-g),$$

where g is the average fraction of secondary electron energy lost in radiative interactions (bremsstrahlung and $\beta$ + annihilation). For low $Z$ and photons energy <1 MeV, $g \rightarrow 0$.

## 4.3 Dosimetric Quantities

Dosimetric quantities needed to assess radiation doses are based on the measure of the energy deposited by radiation in a target. They are absorbed dose, kerma and cema, and the formerly quantity known as exposure.

Absorbed dose describes the energy imparted to matter by all kinds of ionizing radiation in any irradiation geometry. It is used in radiation biology, clinical radiology, and radiation safety. Kerma and cema are intermediate dosimetric quantities used for theoretical and practical aspects of radiometric measurements.

The absorbed dose is defined as the quotient of mean energy, $d\bar{\varepsilon}$, imparted by ionizing radiation to matter of mass $dm$:

$$D = \frac{\mathrm{d}\bar{\varepsilon}}{\mathrm{d}m}, \quad \text{unit: J.kg}^{-1},$$

where $\mathrm{d}\bar{\varepsilon}$ is the mean energy imparted by ionizing radiation to matter in a volume element and $dm$ is the mass of matter in the volume element. The unit of absorbed dose is joule per kilogram ($\mathrm{J.kg}^{-1}$) and its special name is gray (Gy): 1 Gy = 1 J. $\mathrm{kg}^{-1}$. The former unit of absorbed dose, *rad,* was replaced by gray (Gy) 40 years ago and its use is not recommended today. The correlation existing between them is 1 rad = $10^{-2}$ Gy. Absorbed dose is a measurable quantity and primary standards exist to determine its value.

The energy imparted $\bar{\varepsilon}$ is the sum of the energies of all charged and uncharged particles entering the volume of interest minus all the energy leaving the volume, taking into account any mass–energy conversion resulting from interactions or radioactive decay within the volume.

Dose rate is the amount of radiation absorbed per unit time and its unit is $J.kg^{-1}.s^{-1}$.

Kerma, originally an acronym for "*kinetic energy released in matter*", is a quantity applicable to indirectly ionizing particles such as photons and neutrons, defined as the quotient of $dE_{tr}$ by $dm$

$$K = \frac{dE_{tr}}{dm}, \quad \text{unit: } J.kg^{-1},$$

where $dE_{tr}$ is the sum of the initial kinetic energies of all charged particles liberated by uncharged particles in a material of mass $dm$. The unit of kerma is joule per kilogram ($J.kg^{-1}$) and its special name is gray (Gy). 1 Gy = 1 $J.kg^{-1}$.

Kerma must be defined with respect to a specific material in which the interaction occurs (e.g., air kerma, water kerma, etc.). It can also be defined with respect to the specific material in a given medium (e.g., soft tissue kerma in surrounding water).

Under charged particle equilibrium conditions, the air kerma (in gray) is numerically approximately equal to the absorbed dose in air (in gray).

For monoenergetic photons, kerma is related to the energy fluence by

$$K = \Psi \cdot \left(\frac{\mu_{\Delta}}{\rho}\right)_{E,Z}, \quad \text{unit: Gy},$$

where $\mu_{\Delta}$ is the linear energy-transfer coefficient and $\mu_{\Delta}/\rho$ is the mass energy-transfer coefficient–function of the photon energy E and atomic number $Z$ of the medium.

Cema is the acronym for "*converted energy per unit mass*". It is a quantity applicable to directly ionizing radiation such as electrons and protons. Cema is defined as the quotient of $dE_c$ by $dm$, where $dE_c$ is the energy lost by charged particles, except secondary electrons, in collisions in a material of mass $dm$:

$$C = \frac{dE_c}{dm}$$

The unit of cema is joule per kilogram ($J.kg^{-1}$) and is special name is gray (Gy). 1 Gy = 1 $J.kg^{-1}$.

Exposure ($X$) as a physical quantity—not as "to be exposed to"—is the oldest dosimetric quantity originally defined as a measure of the strength of a radiation field at some point in air. It is conceptually limited to the ionization caused by photon radiation ($X$-rays and gamma radiation) in air and is expressed as the quotient of $dQ$ by $dm$:

$$X = \frac{dQ}{dm}, \quad \text{unit: } C.kg^{-1},$$

where d$Q$ is the absolute value of the total charge of the ions of one sign produced in air when all the electrons and positrons liberated or created by photons in air of mass d$m$ are completely stopped in air.

The unit of exposure until 1972 was the roentgen (R); in the SI system, the unit of exposure is coulomb per kilogram ($C.kg^{-1}$). 1 R = 2.58 $\times$ $10^{-4}$ $C.kg^{-1}$.

Exposure has been replaced by air kerma and presently the term is only used to define the act or condition of being subject to irradiation. Exposure should not be used as a synonym for dose because dose is a measure of the effects of exposure [1]. The relation between total kerma and exposure is obtained by

$$K_{\text{air}} = X \cdot \left(\frac{W_{\text{air}}}{e}\right) \cdot \frac{1}{1-\bar{g}},$$

where $W_{\text{air}}$ is the average energy expended in air per ion pair formed, which best estimate is 33.97 $\times$ 1.602 $\times$ 1019 J/ion pair; e is 1.602 $\times$ $10^{-19}$ C/ion pair; and $\bar{g}$ is the radiative fraction, i.e., the average fraction of the kinetic energy of secondary charged particles that is subsequently lost in radiative (photon-emitting) energy-loss processes as the particles slow to rest in the media; the higher the energy, the larger $\bar{g}$.

## 4.4 Special Quantities

Radiological protection is concerned with controlling exposures to ionizing radiation so that tissue reactions are prevented and the risk of stochastic effects is limited to acceptable levels [3]. Tissue reactions are not seen at lower doses like those present in occupational exposures, but it is supposed that damage to the genetic material may occur, which can result in an increase in the risk of cancer observed years later, or heritable disease in future generations. To demonstrate compliance with exposure limits, the ICRP and the International Commission on Radiation Units and Measurements (ICRU) have introduced special quantities in addition to the absorbed dose. These quantities are based on measures of the energy imparted to organs and tissues of the human body. They allow quantification of the extent of exposure to ionizing radiation from both whole and partial body irradiation from external radiation sources and from intakes of radionuclides.

Special quantities are used to manage and limit the radiation risk to both occupational exposure and public exposure; use the absorbed dose as the fundamental physical quantity, average it over specified organs and tissues, and apply suitably chosen weighting factors to take account of differences in biological effectiveness of different radiations and the differences in radiation sensitivities of organs and tissues to stochastic health effects. Special quantities are applicable only for radiation safety purposes within the range from zero to <100 mSv; are based

upon the assumption of a linear, non-threshold, dose-response relationship (LNT); and allows the addition of doses from internal[1] and external[2] exposure [3].

The quantity equivalent dose is related to an averaged effect in an organ/tissue for different types of radiations and is defined by

$$H_T = \sum_R w_R \cdot D_{T,R}, \quad \text{unit: J.kg}^{-1},$$

where $D_{T,R}$ is the mean absorbed dose in the volume of a specified organ or tissue, $T$, due to the radiation of type $R$, and $w_R$ is the radiation weighting factor for radiation $R$. The sum is performed over all types of radiations involved. The unit of equivalent dose is J.kg$^{-1}$ and has the special name Sievert (Sv).

The mean absorbed dose in a tissue/organ $T$ is defined by

$$D_T = \frac{\varepsilon_T}{m_T}, \quad \text{unit: J.kg}^{-1},$$

where $\varepsilon_T$ is the mean total energy imparted in a tissue/organ $T$ and $m_T$ is the mass of that tissue or organ.

In a region of an organ/tissue $T$ the mean absorbed dose is defined by

$$\bar{D}_T = \frac{\int_T D(x,y,z) \cdot \rho(x,y,z) \cdot \mathrm{d}V}{\int_T \rho(x,y,z) \cdot \mathrm{d}V}, \quad \text{unit: J.kg}^{-1},$$

where $V$ is the volume of the region $T$; $D$ the absorbed dose at a point $(x, y, z)$ in that region; and $\rho$ the mass density at this point.

The extent to which the average dose over an organ, tissue, or tissue region is representative of the absorbed dose depends on the homogeneity of the exposure and, for external radiation, on the radiation incident on the body. In cases of extreme partial body exposure, tissue damage may occur even if the mean organ/tissue dose is below the dose limit. For radiations emitted by radionuclides from an intake, the absorbed dose distribution in organs will depend on the specific amount of radionuclide in each organ/tissue and its radiation. Thus, the absorbed dose distribution for radionuclides emitting alpha particles, soft beta particles, low energy photons, or Auger electrons may be highly heterogeneous [3].

The revised set of $w_R$ values most recently adopted by ICRP [3] is shown in Table 4.1. These reviewed values are based on the results of a broad range of

[1]Internal exposure means that an intake of radionuclides has occurred via ingestion, inhalation, through wounds or the skin or by direct injection. Depending on its nature, the source of exposure is transported by and retained in different organs or tissues.

[2]External exposure means that the source of exposure is outside the body and the emitted radiation incident on it.

**Table 4.1** Radiation weighting factors ["From ICRP Publication 103, 2007, with permission of the ICRP."] (All values relate to the radiation incident on the body or, for internal radiation sources, emitted from the source (the incorporated radionuclide).)

| Radiation type | Radiation weighting factor, $w_R$ |
|---|---|
| Photons (*X*-rays and gamma rays) | 1 |
| Electrons and muons | 1 |
| Protons and charged pions | 2 |
| Alpha particles, fission fragments, heavy ions | 20 |
| Neutrons | A continuous function of neutron energies: |
| $E_n < 1$ MeV | $w_R = \begin{cases} 2.5 + 18.2 \cdot e^{-[\ln(E_n)]^2/6} \\ 5.0 + 17.0 \cdot e^{-[\ln(2E_n)]^2/6} \\ 2.5 + 3.25 \cdot e^{-[\ln(0.04E_n)]^2/6} \end{cases}$ |
| 1 MeV $\leq E_n \leq$ 50 MeV | |
| $E_n >$ 50 MeV | |

relative biological effectiveness (RBE) data collected from 1990 up to now compared to the effects of *X*-and $\gamma$-rays at low doses.

In view of the strong dependency of biological effectiveness of neutrons on the neutron energy, for neutrons is currently recommended a continuous function for use in calculations instead of tabulated values (see the equation in Table 4.1). The $w_R$ values for protons have been reduced from 5 to 2 due to more information available related to these particles.

The radiation weighting factor $w_R$ is related to the quality factor $Q$ and represents the relative biological effectiveness of the different radiations with respect to stochastics effects. For example, it is well known that high-LET radiations, including neutrons and alpha particles, cause more damage per unit of absorbed dose than low-LET radiations.

On the other hand, the removing of an electron from a core level of an atom may occur after an inner shell excitation and an electron from a higher energy level may replace it, releasing a specific amount of energy which, when transferred to another electron, ejects a second called Auger electron. A radionuclide, which decays via internal conversion, often emits many Auger electrons with energies of few keV. These emissions can result in a high density of energy imparted and the biological effect may, therefore, be similar to that of a high-LET radiation.

But to understand the response of the body to radiation, it is important to also consider the interaction with different cells, organs and tissues. The effective dose, $E$, provides a value based on detriment-adjusted nominal risk coefficients for cancer and hereditary effects. The unit of effective dose is $J.kg^{-1}$ with special name Sievert (Sv).

$$E = \sum_T w_T \cdot H_T, \quad \text{unit: } J.kg^{-1},$$

where $H_T$ is the equivalent dose in the organ/tissue $T$ weighted by radiation; $w_T$ is the tissue weighting factor for organ/tissue $T$ and $\sum w_T = 1$. It is the factor by which the radiation weighted dose in a tissue or organ $T$ is weighted to represent the

**Table 4.2** Tissue weighting factors

| Tissue or organ | $w_T$ | $\sum w_T$ |
|---|---|---|
| Bone marrow (red), colon, lung, stomach, breast, remainder tissues[a] | 0.12 | 0.72 |
| Gonads | 0.08 | 0.08 |
| Bladder, esophagus, liver, thyroid | 0.04 | 0.16 |
| Bone surface, brain, salivary glands, skin | 0.01 | 0.04 |
| Total | | 1.00 |

[a]*Remainder Tissues* Adrenals, Extrathoracic (ET) region, Gall bladder, Heart, Kidneys, Lymphatic nodes, Muscle, Oral mucosa, Pancreas, Prostate (♂), Small intestine, Spleen, Thymus, Uterus/cervix (♀)

relative contribution of that tissue or organ to the total detriment resulting from uniform irradiation of the body. The sum of the tissue weighting factors is unity.

The tissue weighting factors most recently adopted by ICRP [3] are given in Table 4.2. They are sex-averaged and are for the assessment of effective dose for workers, as well as members of the public, including children.

The detriment for determining the tissue weighting factors was modeled as a function of life lost, lethality, and loss of quality of life and most of the parameters in the risk models were estimated using cancer incidence data from the studies of the Japanese atomic bomb survivors. The risk of hereditary disease in the first two generations was also taken into account.

The main changes in $w_T$ factors with respect to Publication 60 [4] are breast (0.12 from 0.05), gonads (0.08 from 0.20), and remainder tissues (0.12 from 0.05). In addition, specific $w_T$ values of 0.01 are now given for the brain and salivary glands, which cancer risk is judged to be greater than that of other tissues. In the case of gender-specific differences in cancer incidence based on relative detriment for the ovary of females, the gender-averaged $w_T$ of 0.08 assigned to the gonads (cancer plus heritable effects) is similar to that of the female ovary (0.036), plus heritable effects (0.039). In this way the ovary of females is judged to be sufficiently protected [3]. In the case of the thyroid, the $w_T$ assigned of 0.05 allow for the high susceptibility of young children, so the difference in detriment between genders is considered in a conservative way.

Related to the remainder tissues, despite the changes in number (14 in total, 13 for each sex) and in the tissues/organs listed, the most important fact is that the so-called "splitting rule" is no longer in use. The sum of the $w_T$ values is always 1 by definition.

Equivalent dose and effective dose are not measurable, and their values are assessed using their relationship to either physical radiation field quantities, e.g., air kerma free in air, or particle fluence, or operational dose quantities. For the calculation of conversion coefficients for external exposure, computational phantoms are used for dose assessment in various radiation fields. For the calculation of dose coefficients from intakes of radionuclides, biokinetic models for radionuclides, reference biological data, and computational phantoms are used.

Computational phantoms are computer models of human anatomy used in the calculation of radiation dose distribution in the human body. Depending on the manner to represent human anatomy they could be stylized and tomographic. Stylized phantoms describe human anatomy using simple mathematical equations of analytical geometry, while tomographic phantoms are based upon three-dimensional imaging techniques, such as magnetic resonance imaging (MRI) and computed tomography (CT). They represent the human anatomy with a large number of voxels that are assigned tissue type and organ identity. The anatomical computational phantoms adopted by ICRP [3] to be used in the calculations for both internal and external exposures are voxel models constructed from medical image data of real people [5]. In these computational phantoms of the male and female human body, organ masses have been adjusted to approximate those assigned to the ICRP Reference Adult Male and Female without compromising their anatomic realism.

When a radionuclide is introduced into the human body by an intake, it is distributed among the different organs and tissues depending on its physical–chemical characteristic, and will irradiate them over time periods determined both by its physical half-life and its biological retention within the organs and/or tissues. For example, $^{3}$H has a long physical half-life (12.3 years), but a short biological half-time (10 days); while $^{90}$Sr ($T½$ 29.1 years), which behaves chemically much like calcium, tends to concentrate in the bones and its biological half-life in this tissue is approximately 50 years.

Committed dose quantities are used to estimate the radiation dose over extended periods of time (see Fig. 4.1). The committed dose from an incorporated radionuclide is the total dose expected to be delivered within a specified time period ($\tau$). The committed equivalent dose, $H_T(\tau)$, in a tissue or organ $T$ is defined by

$$H_T(\tau) = \int_{t_0}^{t_0+\tau} \dot{H}_{\mathrm{T}}(\mathrm{t})\mathrm{dt}, \quad \text{unit: J.kg}^{-1},$$

where $\tau$ is the integration time following the intake at time $t_0$. The commitment periods are 50 years for workers and adult members of the public, and 70 years for infants and children.

Accordingly, the quantity committed effective dose $E(\tau)$ is then given by

$$E(\tau) = \sum_T w_T H_T(\tau), \quad \text{unit: J.kg}^{-1}$$

The quantity collective effective dose is used to evaluate the dose to a group of occupationally exposed individuals, a local population, or any group of individuals, and with the purpose of only comparing, e.g., radiological technologies and protection procedures, for radiation exposure optimization. Collective effective dose is not intended as a tool for epidemiological studies, and it is inappropriate to use it in

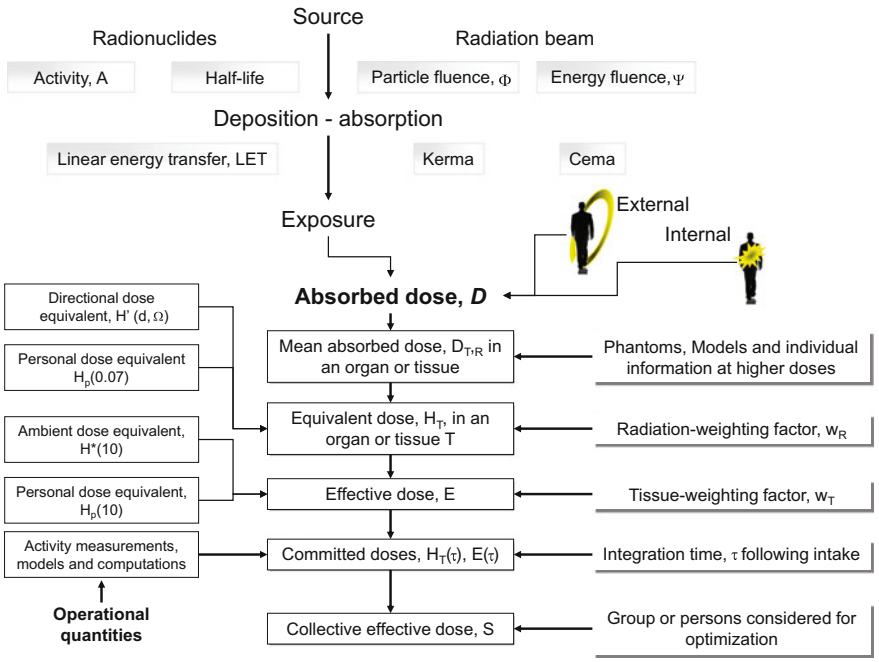



**Fig. 4.1** System of quantities for radiological protection

risk projections. Collective effective dose is defined for a specific dose range from $E_1$ to $E_2$, and specific time period, $\Delta T$, as:

$$S(E_1, E_2, \Delta T) = \int_{E_1}^{E_2} E \frac{\mathrm{d}N}{\mathrm{d}E} \mathrm{d}E, \quad \text{unit: man Sv},$$

where $\frac{\mathrm{d}N}{\mathrm{d}E}$ denotes the number of individuals who are exposed to an effective dose between $E$ and $E + \mathrm{d}E$, and $\Delta T$ specifies the time period within which the effective doses are summed. The special name of the unit of collective effective dose is the man Sievert (man Sv).

## 4.5 Operational Quantities

Since equivalent dose and effective dose cannot be measured directly in the body and the limits are given in these quantities, operational measurable quantities, along with models and computations, have been introduced in the system of radiological protection to assess the dose.

The operational quantities for radiation monitoring in situations of external exposure are ambient dose equivalent $H^*(10)$—for area monitoring—and personal dose equivalent $H_p(10)$—for individual monitoring. Both quantities are designed to control the effective dose. To control the dose to the skin, the hands and feet, and the lens of the eye, the operational quantities are directional dose equivalent $H'(d, \Omega)$—for area monitoring—and personal dose equivalent $H_p(0.07)$—for individual monitoring.

Area monitoring measurements are performed free in air for controlling the workplaces and for defining controlled or restricted areas, while personal dosimeters are worn at the body, where the radiation field is strongly influenced by radiation backscatter and absorption in the body. The operational quantity used in this case takes this situation into account.

The ambient dose equivalent, $H^*(10)$, at a point in a radiation field, is the dose equivalent that would be produced by the corresponding expanded and aligned field[3] in the ICRU sphere[4] at a depth of 10 mm on the radius vector opposing the direction of the aligned field.

The directional dose equivalent, $H'(d, \Omega)$, at a point in a radiation field, is the dose equivalent that would be produced by the corresponding expanded field in the ICRU sphere at a depth, $d$, on a radius in a specified direction $\Omega$. For low-penetrating radiation, it is d = 0.07 mm and $H'(d, \Omega)$ is then written as $H'(0.07, \Omega)$.

The personal dose equivalent, $H_p(d)$, is the dose equivalent in ICRU (soft) tissue at an appropriate depth, $d$, below a specified point on the human body. The specified point is usually given by the position where the individual dosimeter is worn. For the assessment of effective dose, a depth $d$ = 10 mm is recommended, and for assessing equivalent dose to the skin, and to the hands and feet, a depth $d$ = 0.07 mm. For a dosimeter position in front of the trunk, the quantity $H_p(10)$ mostly furnishes a conservative estimate of the effective dose, even in cases of lateral or isotropic radiation incidence on the body.

ICRU stated that $H^*(10)$ and $H_p(10)$ are designed for monitoring strongly penetrating radiation, e.g., photons (above about 12 keV) and neutrons, while $H'(0.07, \Omega)$ and $Hp(0.07)$ are applied for monitoring low-penetrating radiation, e.g., beta particles. It is also used for monitoring the doses to the hands and feet from all ionizing radiation.

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[3] An expanded radiation field is an hypothetical radiation field in which the spectral and the angular fluence have the same values in all points of a sufficiently large volume equal to the values in the actual field at the point of interest. The aligned and expanded radiation field is obtained if all radiation is aligned in the expanded radiation field so that it is opposed to a radius vector Ω specified for the ICRU sphere.

[4] It is a sphere, 30 cm in diameter of tissue-equivalent material with density 1 g $cm^{-3}$ and mass composition: 76.2 % oxygen, 11.1 % carbon, 10.1 % hydrogen, and 2.6 % nitrogen (ICRU soft tissue). It adequately approximates in most cases the human body as regards the scattering and attenuation of the radiation fields under consideration.

For intakes of radionuclides, the first step is to assess the activity of intake either from direct measurements (e.g., measuring the radioactivity of the whole body by whole body counter or of specific organs and tissues by external counting devices) or indirect measurements (e.g., measuring the activity rate in excreta samples, or the activity concentration in the air), and then to apply biokinetic and dosimetric models to calculate the effective dose using reference dose coefficients (doses per unit intake, Sv $Bq^{-1}$). Dose coefficients for the estimation of the committed effective dose for ingestion and inhalation of radionuclides by workers and members of the public are given in Tables III-2A–III-2H of the International Basic Safety Standards [6]. Dose Coefficients based on ICRP Publication 60 [4] are found in ICRP Publication 119 [7].

Dose Coefficients for Intakes of Radionuclides via Contaminated Wounds for 38 radionuclides based on NCRP Wound Model and ICRP biokinetic models can be found at the ORISE website (Oak Ridge Institute for Science and Education) [8]; Conversion Coefficients for Radiological Protection Quantities for External Radiation Exposures, in ICRP Publication 116 [9].

Since exposures of workers may arise from external and internal radiation sources, the effective dose for occupational exposure in most situations can be derived from operational quantities by the following formula:

$$E \cong H_p(10) + E(50), \quad \text{unit: J.kg}^{-1}\text{(Sv)},$$

where $H_p(10)$ is the personal dose equivalent from external exposure (which value is usually obtained by personal dosimeters) and $E(50)$, the committed effective dose from internal exposure over 50 year:

$$E(50) = \sum_j e_{j,inh}(50) \cdot I_{j,inh} + \sum_i e_{j,ing}(50) \cdot I_{i,ing}, \quad \text{unit: J.kg}^{-1},$$

where $e_{j,inh}(50)$ is the committed effective dose coefficient for activity intakes by inhalation of a radionuclide $j$, $I_{j,inh}$ is the activity intake of a radionuclide $j$, by inhalation; $e_{j,ing}(50)$ is the committed effective dose coefficient for activity intakes of a radionuclide $j$ by ingestion, and $I_{j,ing}$ is the activity intake of a radionuclide $j$ by ingestion. For compliance with dose limits and management of staff, the estimated committed dose is assigned to the year in which the intake occurred.

The effective dose for occupational exposures is called “dose of record” and is used to demonstrate compliance with dose limits.

In the calculation of the effective dose from specific radionuclides, it is also important to take into account the specific characteristics of the material into the body, including the activity medium aerodynamic diameter (AMAD) of the inhaled aerosol, and the chemical form of the particulate matter to which the specified radionuclide is attached.

The Annual Limit on Intake (ALI) and the Derived Air Concentration (DAC) from 1991 are old derived parameters that may be useful in the control of exposures. Both concepts can help in various practical situations, e.g., in characterizing

the relative hazard of radiation sources to ensure that appropriate administrative controls are in place. The ICRP does not now give any ALI or CDA values.

The ALI was defined as an intake of a radionuclide $j$ (in Bq) which would lead to a committed effective dose of 20 mSv, i.e., the average annual limit on effective dose for workers $E_{limit,w}$, in mSv under the assumption that the workers were exposed through that pathway only:

$$\mathrm{ALI}_j = \frac{E_{limit,w}}{\mathrm{e}(50)}, \quad \text{unit: Bq},$$

where $e(50)$ is the corresponding committed effective dose coefficient in $\mathrm{mSv.Bq^{-1}}$.

The DAC was defined as the activity concentration in air of the radionuclide $j$ which would lead to an intake of an ALI (in Bq) assuming a gender-averaged breathing rate of 1.1 $\mathrm{m^3\ h^{-1}}$ and an annual working time of 2000 h.

$$\mathrm{DAC}_j = \frac{\mathrm{ALI}_j}{2200\ \mathrm{m^3}}, \quad \text{unit: Bq.m}^{-3}$$

The DAC for inert gases, which are not incorporated, is limited by the effective dose arising from radiations incident on the body from the airborne activity. Thus the DAC is given by

$$\mathrm{DAC} = \frac{E_{\mathrm{lim},w}}{2000\dot{\mathrm{e}}_{\mathrm{sub}}},$$

where $\dot{\mathrm{e}}_{\mathrm{sub}}$ is the effective dose rate coefficient ($\mathrm{mSv\ m^3(Bq\ h)^{-1}}$) for submersion in an airborne cloud containing the noble gas radionuclide and 2000 h is the annual working time. For some radionuclides, the DAC is limited by the dose to the skin.

Figure 4.1 summarizes the most important quantities used in radiation safety, the additional information required, and their relationships.

## References

1. International Atomic Energy (2007) IAEA safety glossary, terminology used in nuclear safety and radiation protection—2007 Edition. IAEA, Vienna
2. NIST (2014) NIST physical measurement laboratory, physical reference data, X-Ray mass attenuation coefficients. The National Institute of Standards and Technology (NIST). http://www.nist.gov/pml/data/xraycoef/index.cfm. Accessed 26 March 2015
3. ICRP (2007) The 2007 recommendations of the international commission on radiological protection. Ann ICRP 37:2–4 April-June 2007
4. ICRP Publication 60 (1991) 1990 recommendations of the international commission on radiological protection. Ann ICRP 21(1–3)
5. ICRP Publication 110 (2009) Adult reference computational phantoms. Ann ICRP 39(2)
6. International Atomic Energy (2014) Radiation protection and safety of radiation sources: international basic safety standards, IAEA safety standards series GSR part 3. IAEA, Vienna

7. ICRP Publication 119 (2012) Compendium of dose coefficients based on ICRP publication 60. Ann ICRP 41(1)
8. Oak Ridge Institute for Science and Education (2015) Dose coefficients for intakes of radionuclides via contaminated wounds. ORAU for US Department of Energy. https://orise.orau.gov/reacts/resources/retention-intake-publication.aspx. Accessed 4 April 2015
9. ICRP Publication 116 (2010) Conversion coefficients for radiological protection quantities for. Ann ICRP 40(2–5)

# Chapter 5
# Measuring Instruments and Methods

Diverse methods and instruments are available for detecting and measuring the presence of radiation. For instance, dose rate meters and survey meters are suitable for measuring ambient dose equivalent; dosimeters to measure individuals' dose equivalent; surface contamination meters to check personnel, equipment, and facilities for radioactive contamination; airborne contamination meters and gas monitors to measure the activity in air; and in vivo counting for monitoring the activity and distribution of contaminants inside the body.

But no simple device can detect all kinds of radiation and not one device is useful in all situations. The type of radiation (alpha, beta, gamma, and neutron), the level of radiation (from background to high level), and the energy resolution (kV, MV) have to be considered, among other features, when selecting the proper measurement device. Furthermore, instruments, which are used to measure ambient dose equivalent, shall have an isotropic response. Instruments, which are used to measure directional dose equivalent and personal dose equivalent, shall have a defined directional response.

Measurements are done indirectly by detecting the effect of ionizing radiation in the detector material. There are two types of instruments to measure radiation—particle counters and dose measuring instruments—and both are based on particle counting. Counts are the number of events detected, but the response of a dose measuring instrument should be proportional to the amount of energy absorbed by the detector. The conversion calculation is dependent on the radiation energy levels, the type of radiation being detected, and the radiometric characteristic of the detector. Results are displayed in units of dose (Gy, Sv) or dose rate (Gy/min, mSv/h). Sometimes the same instrument is capable of both types of readings.



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## 5.1 Common Counting Methods for Monitoring

The radiation counting detector can either be a gas, solid, or liquid; even an emulsion. The measuring principle is based on the interactions of charged particles with the detector leading to ionized and excited molecules along the path.

### *5.1.1 Gas Detectors*

Three types of particle counting instruments are equipped with gas filling detectors. They are ionization chambers, proportional counters, and Geiger–Müller counters. The basic theory of operation relies on the creation of ion pairs (positively charged ions and electrons) in the filling gas by radiation, which are collected and converted into an electrical signal (current or pulse) by the application of an external electrical field.

In ionization chambers no secondary particles are formed—the charges collected are only those produced in the initial ionization event; in proportional counters secondary particles are formed, but their number is proportional to the initial energy of radiation; in Geiger–Müller counters secondary particles are produced in large numbers and the number of ions is no longer proportional to the initial energy. Pulse size strongly depends on the voltage applied to the detector and serves to distinguish the three detector types and their electronic circuits. See how the number of electrons collected through a gas-filled detector varies as applied voltage is increased in Fig. 5.1.

An ionization chamber consists of a chamber filled with gas and a pair of electrodes to collect the created electric charges. As shown in Fig. 5.2, it resembles a cylindrical condenser with a central anode and an electrically conductive wall. A specifically relatively low voltage is applied between electrodes to reduce recombination of the original pairs to an acceptable value (region I in Fig. 5.1) and to create an electric field for sweeping ions to the oppositely charged electrode. Air is the most common filling gas.

When the gas between the electrodes is ionized by radiation, primary ion pairs are formed and, under the influence of the electric field, positive ions and dissociated electrons move to the electrode of the opposite polarity, thus creating an ionization current which may be measured by an electrometer.

Each ion pair created deposits or removes a small electric charge to or from an electrode, such that the accumulated charge is proportional to the number of ion pairs created, and hence the radiation dose. This continual generation of charge produces an ionization current, which is a measure of the total ionizing dose entering the chamber.

The associated electronic is highly sophisticated and capable of measuring very small currents. Ionization chambers can operate with different gas fillings and pressures. These instruments are designed to provide an accurate measure of

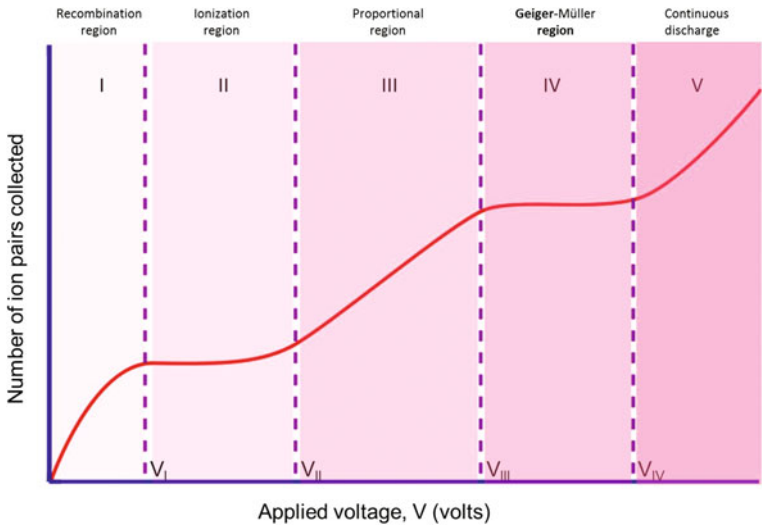


**Fig. 5.1** Relationship between the voltage applied and charge collected

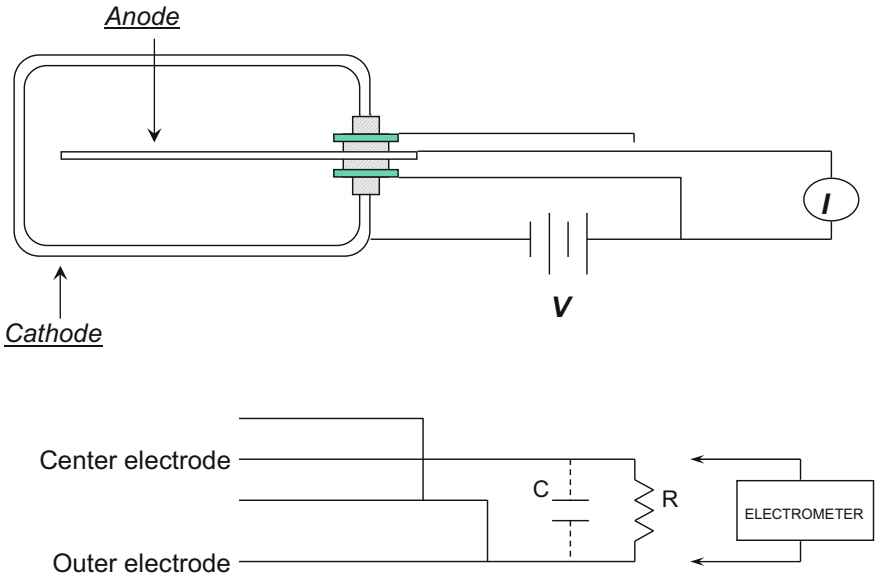


**Fig. 5.2** Ionization chamber simplified scheme

absorbed dose to air which, through appropriate conversion factors, can be related to dose to tissue.

Properly calibrated, ionization chambers have found a wide range of application; in addition to area monitoring and shielding assessment, they are used for

dosimetric measurements in radiation therapy and diagnostic radiology; to measure the activity administered to patients in nuclear medicine; and in industry, in traversing measuring systems and in smoke detectors. These instruments are capable of detecting gamma and *X*-rays, and electrons energetic enough to penetrate the detector wall. To help the entry of less penetrating radiation, the detector is provided with a thin window.

Proportional counters work with voltages increased beyond the ionization chamber region (see Fig. 5.1), and, for that reason, formed primary electrons are accelerated and gain sufficient energy to cause further ionization and produce secondary electrons. In proportional counters, the number of electric charges collected by the anode in a short period of time will be much greater than, but proportional to, the number of primary electrons; and the amplitude of the resulting pulse of current will be proportional to the incident radiation energy. Proportional counters measure the number of ionizing events, whereas ionization chambers measure the amount of ionization produced by these events.

With the help of suitable electronics, proportional counters are capable of identifying and measuring radiations of high LET in the presence of others with lower LET, i.e., can discriminate between alpha and beta particles. They are also used for neutron detection (filled with $BF_3$ or $^3He$), and for *X*-ray spectroscopy to some extent. The main limitation is that they require very stable electronics, gas supply, and technical conditions to ensure constant operation thus, usually, are only used in a laboratory setting. Large-area gas flow proportional counters are used, e.g., in whole body and hand and shoe contamination monitors.

Geiger–Müller counters are one of the oldest devices to measure ionizing radiation, but are still one of the most sensitive, especially for low radiation levels. In Geiger–Müller (GM) counters, the operating voltage between the anode and the cathode is even higher, usually in the 900–1200 V range)—the optimum operating voltage will be about the middle of the plateau (see Fig. 5.1). Like in proportional counters, the higher voltage accelerates the electrons causing further ionization in the gas. However, this cascading of ion pairs occurs to a much larger degree and continues until the counter is saturated with ions. The result is an electrical current pulse of large voltage that is easily counted without further amplification.

Pulse size and width do not depend on the incident radiation energy; thus GM counters can be used for all kinds of radiation. The percentage of the incoming radiation that is counted is known as the efficiency of the tube. As a general rule, a GM tube will give a pulse for each alpha or beta particle entering it—the efficiency of detection is 100 % for $\alpha$, and nearly 100 % for $\beta$—but only 1 or 2 % for $\gamma$ radiation. It is due to a high probability of $\gamma$ photon passing through the sensitive volume without any interaction. GM counters made of different materials and sizes have different efficiencies. Generally, smaller tubes of the same material have lower efficiency.

Geiger–Müller pulses are relatively large and can be easily handled with simple electronics as basically shown in Fig. 5.3. A GM tube connected to a portable count ratemeter with an audible count rate indicator is the most common type of survey meter for contamination and area monitoring. For contamination monitoring, the detector has a thin window and a large detector area (pancake type).

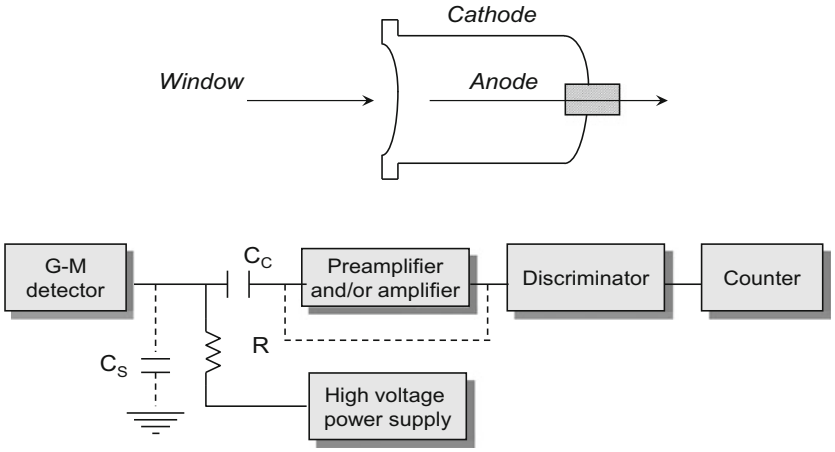


**Fig. 5.3** Geiger–Müller detector simplified scheme

An energy-compensated GM tube is necessary to measure dose rate, so that the dose displayed acceptably relates to the counts detected.

### *5.1.2 Scintillation Detectors*

Scintillation detectors are based on the excitation effect of radiation on certain solid or liquid materials that causes them to emit photons, mostly in visible and ultraviolet regions of the spectrum. In other words, scintillation counters use a material —liquid or solid—also called phosphor, whose atoms are easily excited by ionizing radiation and emit light when returning to their ground state. The output pulse for each photon detected carries information about the energy of the original incident radiation on the scintillator. Thus both intensity and energy of the radiation can be measured.

As shown in Fig. 5.4, the scintillator material—phosphor—is attached to a photomultiplier tube to convert the optical signal into an electric signal. The photon–electron conversion takes place in a thin material called photocathode. Formed electrons are then repeatedly accelerated toward several dynodes, where a large number of secondary electrons are released. The output pulse is then amplified to be measured by the associated electronics. Pulse heights can be measured with appropriate electronics, and when plotting the relative counting rate versus energy it is possible to obtain a spectrum of the source.

Common materials used as scintillator are inorganic phosphors, e.g., ZnS(Ag) (zinc sulfide doped with silver powder), NaI(Tl) (thallium doped sodium iodide crystal), and CsI(Na) (cesium iodide crystal doped with sodium), and organic

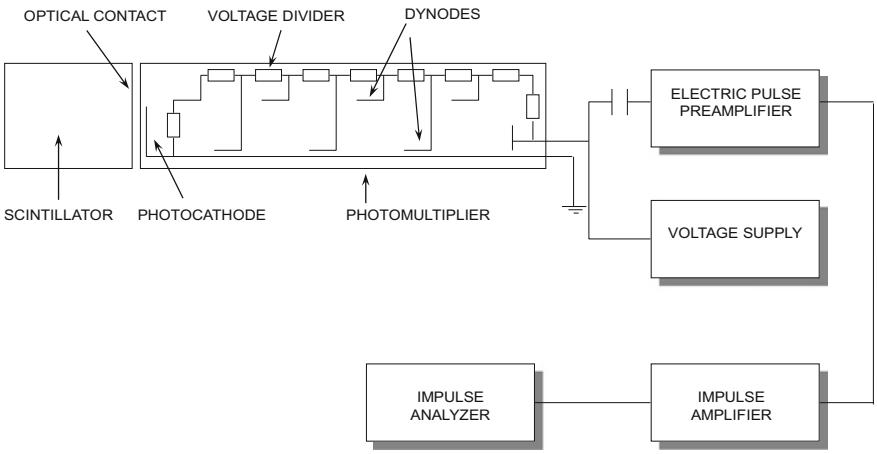


**Fig. 5.4** Basic scheme of a scintillation counter

phosphors, e.g., pure crystals of anthracene, stilbene, and naphthalene, and certain plastics as *p*-terphenyl in toluene (liquid), and *p*-terphenyl in polystyrene (plastic). Scintillation counters have a very good time resolution, and are more sensitive than Geiger–Müller counters, mainly because of the higher density of the detecting medium; they can be used to detect alpha, beta, gamma, and *X*-rays and for $\gamma$ spectroscopy.

Liquid scintillation counting (LSC) is a laboratory measurement method for low-energy beta particles, alpha particles, and Auger electrons emitted by some radionuclides. It involves dissolving the sample to be counted directly into the liquid scintillator. Problems like self-absorption, attenuation of particles by detector windows, and beta backscattering from the detector are completely avoided in liquid detectors.

LSC has eliminated many of the problems associated with $^{14}C$ and $^{3}H$ detection in biological samples. Since the sample is practically mixed into the liquid scintillator—a cocktail involves a solvent, a primary scintillator, and a wavelength shifter—beta particles immediately excite thousands of scintillator molecules, which quickly reemit the absorbed energy in the form of photons traveling freely through the transparent scintillator to the photomultiplier tube (PMT). To detect as much light as possible from a liquid scintillation vial, the sample is typically viewed by two opposed PMT connected in a coincidence circuit. The coincidence circuit assures that only genuine light pulses, which reach both PMT, are counted. The scintillation solvent is benzene ($C_6H_6$) or a mixture of benzene and toluene ($C_6H_6CH_3$). The most common primary scintillator is PPO (2, 5-diphenyloxazole). The next common primary scintillator is butyl PBD [2(4-Biphenyl)-5-(4-tert-butylphenyl)-1, 3, 4-oxadiazole]. The wavelength shifter is the secondary scintillator, which absorbs the fluorescence energy of the excited primary scintillator, and reemits the energy as a longer wavelength signal. The most common secondary scintillator is Bis-MSB [*p*-bis-(o-MethylStyryl)-Benzene].

### 5.1.3 Solid-State Detectors

Solid-state detectors—also called semiconductor radiation detectors—are detectors made of a semiconductor material such as a silicon or germanium crystal. Semiconductors directly convert the incident energy into electrical pulses; hence they provide the highest resolution obtainable and are suitable for accurate measurements of energy and high precision dosimetry. The associated electronics separate the pulses into channels according to the pulse heights resulting in very reliable multichannel analyzers.

Solid-state detectors are also based on the ionization of the detector material, but instead of ion pairs, electron–hole pairs are created in the crystal, and the subsequent movement and collection of charges gives rise to an electrical pulse or current. The advantage of a semiconductor is that the average energy required for creating an electron–hole pair is 10 times smaller than that required for gas ionization; the amount of ionization produced for a given energy is an order of magnitude greater as well.

Semiconductors can be intrinsic and extrinsic. Intrinsic semiconductors are extremely pure; an intrinsic semiconductor must have no more than one impurity atom in 10 billion semiconductor atoms. Extrinsic semiconductors are doped by adding impurity atoms to pure materials to significantly increase its conductivity.

It is possible to increase the number of negative charge carriers in a semiconductor crystal by doping it with an electron donor like phosphorus (P). Electron donors are also known as *n*-type dopants. It is also possible to introduce an impurity lacking an electron, for example, boron (B). This leaves an empty spot in the semiconductor crystal known as a hole, a positive charge carrier. Doping with an electron acceptor creates an excess of holes which can accept electrons. An electron acceptor dopant is also known as a *p*-type dopant. Doped crystalline lattices are schematically shown in Fig. 5.5.

When large numbers of atoms are close to each other like in a semiconductor crystal, available energy levels form a nearly continuous band wherein electrons may move. It is the width of these bands and their proximity to existing electrons that determines how mobile those electrons will be when exposed to an electric field. The band containing electrons is called valence band, and the next band, which is empty, is called conduction band.

Common semiconductor detectors are based on semiconductor junction diodes. They are formed from extrinsic semiconductors. As shown in Fig. 5.6, a junction is an interface where *n*- and *p*-type semiconductors are brought together forming a single system. When it is placed under reverse bias (+ voltage to *n*-type and − voltage to *p*-type) it behaves like a solid ionization chamber. The passage of ionizing radiation through the depletion region (a region with no excess of holes or electrons) makes it thicker and creates electrons in conduction band and holes in valence band. Electrons migrate to positive charges in n-side and holes, to negative voltage on *p*-side, causing an electrical output proportional to the number of electron–hole pairs or energy deposited in the detector. The resulting signal is then amplified and filtered, and a final comparator distinguishes between signal and noise.

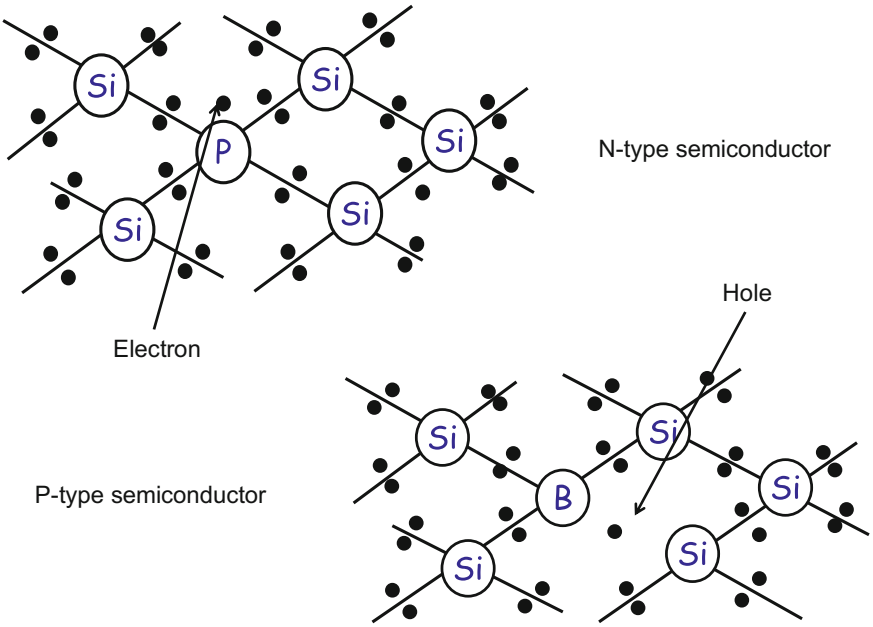


**Fig. 5.5** *N*-type donor impurity (P) creates free electron and *p*-type acceptor impurity (B) creates a hole, a positive charge carrier

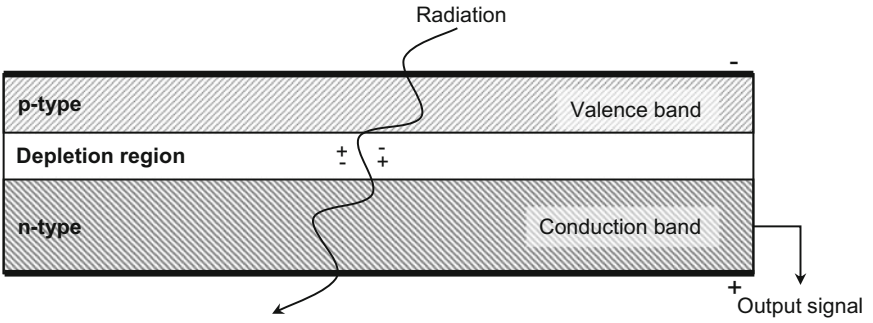


**Fig. 5.6** Enlarged cross section of a semiconductor counter

Bulk conductivity detectors are formed from intrinsic semiconductors of very high bulk resistivity, for example, CdS (cadmium sulfide), CdSe (cadmium selenide), CdTe (cadmium telluride), and CdZnTe (cadmium zinc telluride). They offer the large sensitive detection volumes that are necessary for particle counting and spectroscopy. They also operate like ionization counters, but with a higher density than gases and a tenfold greater ionization per unit absorbed dose.

Among all direct ionization devices, germanium detectors are the ones which have the highest resolution. In spectrometric applications, peaks obtained with

germanium detectors are almost 100 times narrower than the peaks from a sodium iodide detector. High Purity Germanium (HPGe) detectors are made by highly refining the element germanium and growing it into a crystal. The special property of these detectors is that they conduct current in proportion to the photon deposited energy. They operate at cryogenic temperatures (77 °K) to reduce thermal noise, and also require highly accurate supporting electronics.

Until lately, HPGe detectors were large and expensive laboratory instruments, not very suitable for field use. Thanks to the advances in solid-state electronics and particularly in digital signal processing over the past 10 years, the size, complexity, operating power, and cost of the electronics required to support HPGe detectors have been dramatically reduced. Much more recently, miniature, low-power, high-reliability cryogenic coolers have been developed to replace liquid nitrogen in the cooling of HPGe detectors.

### *5.1.4 Other Detectors*

Ionization chambers can also be made of appropriate liquids. Liquid ionization chambers (LIC) have been developed for accurate measurements of absorbed dose in tissue-like materials. They combine the advantages of air-filled ion chambers and solid-state detectors. The entire ion chamber, except the thin wires connecting the two electrodes, is made from materials of low atomic number and density close to unity. The liquid mixture is matched to give a mass energy-absorption coefficient similar to that of water. They are promising detectors that have been successfully used in small beam dosimetry.

Cherenkov counters are more used in physics research for identifying particles, in conjunction with momentum measurements, e.g., in a tracking chamber. Nevertheless, some beta-emitting isotopes (e.g., $^{32}P$) can be analyzed on an LSC system without using scintillators due to Cherenkov radiation. Cerenkov radiation is the blue glow that you see when you look into a reactor pool. It is emitted whenever charged particles pass through a medium at a velocity greater than that of light in the same medium.

Track detectors, also called solid state nuclear track detectors or SSNTD, are based on the damage caused by particles in a small cylindrical region around its trajectory across the detector. The damage depends on the energy released inside the cylindrical region and forms the so-called latent track. The subsequent etching of latent track leads to the formation of etch pit cones, which can be observed with an ordinary optical microscope. The size and shape of these tracks yield information about the mass, charge, energy, and motion direction of the particles.

The main advantages of these detectors are the detailed information available on individual particles, the persistence of the tracks allowing measurements to be made over long periods of time, and the simple, cheap, and robust construction of the detector. SSNTDs—like CR-39 (allyl diglycol carbonate, ADC), and CN-85 (cellulose nitrate)—have found application in fast neutron dosimetry and in studies of radon concentration.

### 5.1.5 Selection Criteria

An ideal radiation detection system should have a high efficiency and a good energy resolution. Spatial resolution, sensitivity, and dead time are also important factors. In some cases you have to pay attention to sample preparation and analysis conditions as well. When selecting the radiation monitoring instruments for any particular application, the following criteria should be considered [1] ["Reproduced with permission by the IAEA."]:

- The type of radiation to be measured—alpha, beta, gamma, neutrons, *X*-rays;
- The quantity to be measured—dose, dose rate, activity, or contamination;
- The energy response of the instrument;
- Unwanted responses and overload performance—noise, leakage current, off-scale, or overload indication;
- The sensitivity and range of measurements required;
- The speed with which the instrument responds;
- Scales logarithmic/linear analog or digital displays and ease of use;
- Illuminated display and/or audible output;
- Response in ambient temperatures, humidity, radiofrequencies, magnetic fields, etc.;
- Intrinsic safety in explosive/flammable locations;
- Ease of decontamination;
- Battery availability and life expectancy;
- Size, weight, and portability;
- Ruggedness, reliability, and serviceability.

## 5.2 Individual Monitoring and Personal Dosimetry

Individual monitoring includes the assessment of dose from external and internal exposures. Dose from external exposure can be measured with a suitable personal dosimeter, which provides an estimate of the radiation dose deposited in the individual wearing the device. Dosimetry systems widely used for individual monitoring are thermoluminescent dosimeters and film badges. Both are passive systems that have to be processed after the exposure to obtain the dose measurement. The dosimeter typically remains in place for a period of time (1–3 months) to assess the cumulative dose.

Dose from internal exposure is computed by the application of biokinetic and dosimetric models to the results of activity directly measured in organs, the whole body or wounds, or indirectly measured in samples of urine and feces, and/or in air at the workplace [2]. The effective dose is determined from the estimate of the intake using the dose coefficients (effective dose per unit intake) for the radionuclides, as appropriate [3].

### 5.2.1 Film Badge Dosimetry

Photographic emulsions (films) have been used for radiation detection since the discovery of radioactivity and *X*-rays. Photographic film typically consists of a silver halide radiation-sensitive emulsion embedded in gelatin, and coated on a transparent polyester base. When the emulsion is exposed to radiation, excitation, and ionization take place in the silver halide crystals that lead to the formation of a latent image, which is subsequently amplified by the film development and measured as optical density (OD). The relationship between OD and dose is known as sensitometric curve, which should be linear with dose and do not depend on the dose rate and energy.

A film badge dosimeter contains a photographic film and a series of filters in a holder. The film is usually coated with two emulsions. One side is coated with a large grain, fast emulsion that is sensitive to low levels of exposure. It is used to estimate doses from 0.05 to 50 mSv. The other side of the film is coated with a fine grain, slow emulsion that is less sensitive to exposure. The low emulsion is used for doses up to 10 Sv. The emulsions may be of the same or separate bases and are sealed in a special light proof, vapor proof wrap to prevent light, moisture, or chemical vapors from affecting the film.

The specially designed photographic film holder shown in Fig. 5.7 has a system of filters made of different materials to (a) facilitate a homogeneous detector response to various radiation energies; (b) discriminate the type and energy of incident radiation; and (c) detect the direction of incidence of radiation. The badge holder also contains an open window to determine the radiation dose due to beta particles.

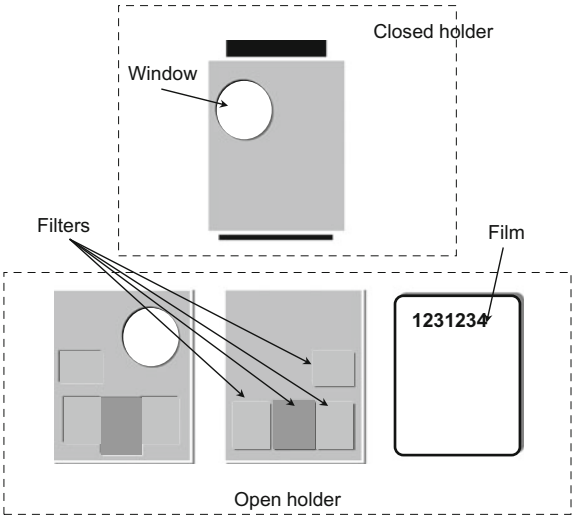


**Fig. 5.7** A film badge dosimeter

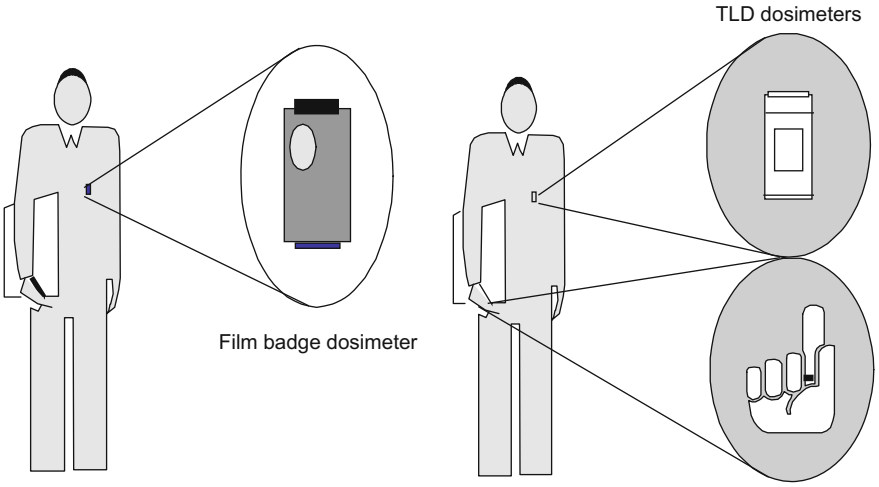


**Fig. 5.8** Dosimeters used for routine individual monitoring

Radiation of a given energy is attenuated to a different extent by various types of absorbers. Therefore, the same quantity of radiation incident on the badge will produce a different degree of darkening under each filter. Thus, cumulative doses from beta particles, *X*-rays, gamma rays, and thermal neutron (aluminum, cadmium, tin, and lead filters) are calculated by measuring the optical densities under the filters and comparing the results with the results of calibration films that have been exposed to known doses.

Film badge dosimeters are not useful to measure alpha particles or low-energy beta particles since these particles cannot pass through the film wrapper. Film badges need to be worn correctly so that the dose they receive accurately represents the dose the wearer receives. Whole body badges are worn on the body between the neck and the waist, fastened to the outside of clothing, often to a shirt pocket and facing forward (see Fig. 5.8).

The major advantages of a film badge dosimeter are that it provides a permanent record that can be reexamined, it is able to distinguish between different energies, and can measure doses due to different types of radiation.

### *5.2.2 Luminescence Dosimetry*

There are some materials that, upon absorption of ionizing radiation, retain part of the absorbed energy and are capable of releasing it in the form of ultraviolet, visible, or infrared light when using an exciting agent. If the exciting agent is heat, it is called a thermoluminescent material or phosphor. If the exciting agent is light, the phenomenon is referred to as photoluminescence and optically stimulated

luminescence (OSL). The amount of light released can be measured and then related to the radiation dose.

Thermoluminescent dosimeters (TLDs) deposit all or part of the energy received from ionizing radiation to their crystal lattice producing free electrons and holes. Due to intentionally introduced impurities in the crystal, electrons jump to higher energy states and stay trapped and locked into place. When the crystal is heated, trapped electrons return to their original ground state and release the captured energy in the form of light. Since the amount of light emitted is very small, the TLD dosimeter is placed in a dark chamber equipped with a photomultiplier tube for reading. The photomultiplier converts the light into an electronic signal which is then amplified. The resulting output of the TLD reader is called a "glow curve" and the area under this curve is directly proportional to the amount of radiation that was absorbed in the dosimeter.

Thermoluminescent dosimeters may vary in design, but they typically consist of two or more thermoluminescent chips (or discs in a polytetrafluoroethylene matrix, or powders), enclosed in a plastic holder (badge or card) as shown in Fig. 5.9.

Some phosphors such as LiF:Ti, Mg (lithium fluoride doped with magnesium and titanium), CaSO4:Dy (dysprosium-doped calcium sulfate), and $CaF_2$:Mn (calcium fluoride doped with manganese) form useful TLDs, but almost all current dosimetry systems only use lithium fluoride (LiF) doped with magnesium and titanium or thallium because it is tissue equivalent, do not require complex filter systems and has a good linear response between 100 μSv and 5 Sv [4]. LiF TLDs are used for monitoring whole body exposure to *X*-rays, gamma rays, and beta particles. Lithium borate ($^{6}$Li and $^{10}$B) can be used together with lithium fluoride to monitor the dose from thermal neutrons.

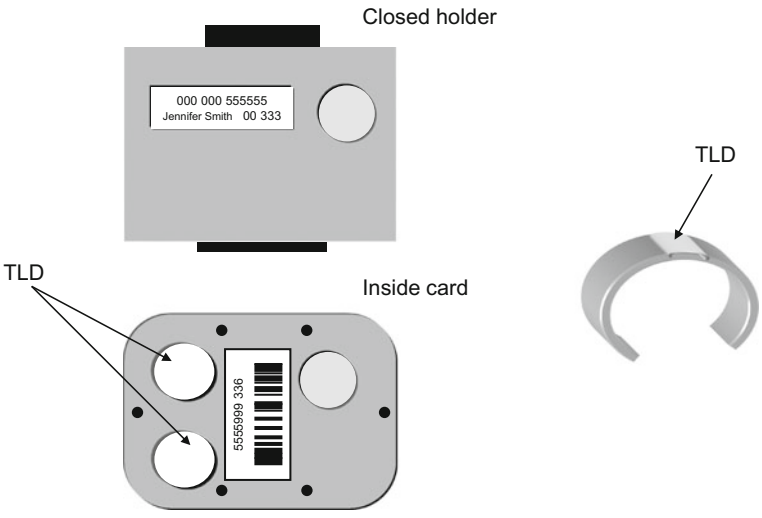


**Fig. 5.9** TLD holder and ring

The advantages of a TLD over other personnel monitors are its linearity of response to dose, its relative energy independence, and its sensitivity to low doses. It is also reusable, which is an advantage over film badges. However, no permanent record is provided. They are less easily damaged than film badges, can be used longer (for 3 months) and are ideal for extremity monitors because of their small size, energy response characteristics, and linearity through a wide range of dose and dose rates. Finger ring dosimeters are also shown in Figs. 5.8 and 5.9.

Radiophotoluminescence (RPL) and optically stimulated luminescence (OSL) are based on physics phenomena similar to TLDs.

Radiophotoluminescent dosimeters (RFLDs) use silver-doped phosphate glass as radiophotoluminescence material. Some glasses, such as the GD-450, the GD-352 M, and the SC-1 have been studied for personal dosimetry, dose measurements in radiology, and environmental radiation dose monitoring, respectively [5].

When the glass is exposed to radiation, electron–hole pairs are produced. Electrons are then captured into $Ag^+$ ions, changing the $Ag^+$ into $Ag^0$. On the other hand, holes are captured by $PO_4$ tetrahedrons at the beginning of the migration producing $Ag^{2+}$ due to their interaction with $Ag^+$ ions. Both $Ag^0$ and $Ag^{2+}$ form stable luminescence centers. Upon excitation with a pulsed UV laser ($N_2$ gas laser), the electrons in color centers jump to higher energy level, emit light, and then return to the original color centers [6]. A photomultiplier registers the orange fluorescence emitted by the glass. Because the electrons return to the electron traps, the signal is not erased during the readout, thus the dosimeter can be reanalyzed several times; the measured intensity is proportional to the amount of radiation.

Typically RPLDs cover the dose range from 30 μSv to 10 Sv and have a flat energy response within 12 keV to 8 MeV [7]. Accumulation of the dose is also possible. RPL signal exhibits very low fading and is not sensitive to the environmental temperature.

Optically stimulated luminescence (OSL) dosimeters use a thin layer of aluminum oxide doped with carbon ($Al_2O_3$:C) [8]. Likewise TLDs and RPLDs, these dosimeters make use of electrons trapped between the valence and conduction bands in the crystalline structure. The trapping sites are the impurities of the crystal lattice and the stimulating energy source is light (UV, visible, or infrared). During analysis, the aluminum oxide crystal is stimulated repeatedly with selected frequencies of laser light to provide multiple dose evaluations of luminescence for improved precision. OSL dosimeters can be reanalyzed several times and may be used for up to 1 year.

Commercially available Luxel® dosimeter [9] shown in Fig. 5.10 consists of a thin strip of aluminum oxide ($Al_2O_3$) mounted in a filter pack. The filters include an image filter to help determine the direction of the exposure. This assembly is sealed in a hexagonal plastic blister pack. After laser stimulation, the image of the luminescent pattern is recorded to determine the amount of radiation and the filter patterns are analyzed to determine the type, energy, and quantity of the radiation. Luxel® dosimeter can be used down up to 10 μSv for *X* and gamma rays and 100 μSv for beta particles, and up to 10 Sv in photon beams from 5 keV to 40 MeV.


**Fig. 5.10** Landauer Luxel® radiation monitoring badge ["Reprinted from monitoring of radiation exposure at https://www.uic.edu/depts/envh/RSS/Badges.html with permission of Landauer, Inc."]

### *5.2.3 Other Personal Dosimeters*

Self-reading pocket dosimeters are small ionization chambers filled with air to indicate and measure individual doses in real time. Some units have a built-in charger; others need an external charge prior to be worn. The charge stored is reduced when radiation ionizes the air in the chamber, allowing a direct reading. The dose can be read by looking through the eyepiece on one end of the dosimeter, pointing the other end toward a light source.

Pocket dosimeters detect and measure *X*- and gamma rays from 0.016 to 2 MeV and beta particles above 1 MeV, from 0–2 mSv and 0–50 mSv. They are also sensitive to thermal and fast neutron when the chamber is coated with boron or plastic, respectively. They often present charge leakage problems, but could be useful in special operations, e.g., source replacements, leakage tests, etc., and in handling of radiation incidents or emergencies.

Available modern electronic personal dosimeters (EPD) are more sensitive and reliable than pocket dosimeters. They could be small volume ionization chambers, miniature Geiger–Müller counters, or silicon detectors, featuring complete analog and digital circuitry, nonvolatile EEPRO memories, and audio and visual alarms. They are typically standalone devices, but most of them actually have data communication capability via PC software. EPD are useful for real-time readings in almost all radiation working environments—nuclear power plants, industrial radiography in an open field, a radiotherapy unit, a laboratory using radionuclides, etc.—and for first responders and emergency workers as well. Today some of them can exchange data with user's mobile device in real time via Bluetooth wireless

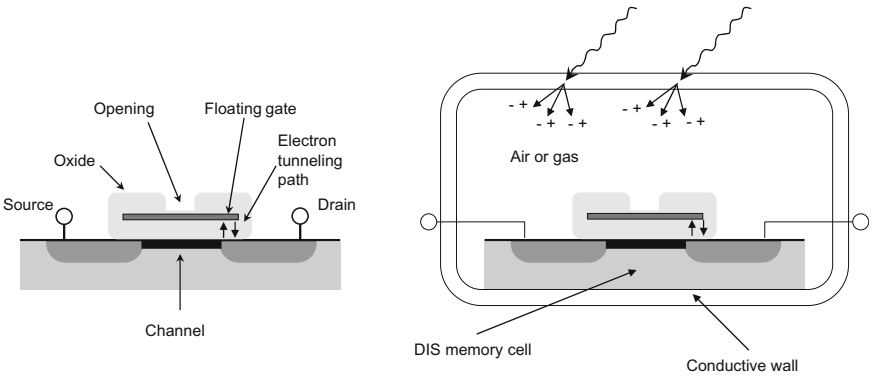


**Fig. 5.11** Work principle of a DIS dosimeter bag ["Reproduced from DIS-1 dosimeter user's guide, with permission of Mirion Technologies (RADOS) Oy"]

connection (iOS, Android). The DOSIMAN Electronic Personal Dosimeters in Credit Card Size—DOSICARD™—from Canberra is the smallest dosimeter available in the market. Ultrathin, compact, lightweight, and shockproof, features LCD display, large nonvolatile memory, and programmable alarm levels on doses from 1 μSv to 10 Sv, and dose rates form 1 μSv/h to 1 Sv/h [10].

The innovative personal dosimeter known as direct ion storage (DIS) consists of a small volume ion chamber housed in a conductive tissue-equivalent wall coupled to a nonvolatile memory cell (EEPROM).

The construction of a DIS memory cell is represented in Fig. 5.11 [11, 12]. Prior to use, electrons have to be tunneled into the floating gate of the nonvolatile solid-state memory cell (EEPROM) through the oxide layer. This creates the potential between the wall of the small chamber and the gate. The oxide layer has an opening allowing the surface of the floating gate to be in direct contact with the air (or any other gas). The ionizing radiation incident in the air or gas produces electron–ion pairs with extremely high mobility, whereas the electric field surrounding the floating gate allows efficiently transfer these charges to the gate before any recombination occurs.

Since the change in current between the source and drain can be measured and related to radiation dose, the instant reading capability of the dosimeter allows the user to observe the accumulated dose on a daily basis using a badge reader; the number of readouts is unlimited. On the other hand, dose information can be stored automatically into a database every time a readout is performed. The DIS dosimeter and its reader are shown in Fig. 5.12. DIS dosimeters are used to measure gamma and *X*-ray doses from 1 μSv to 40 Sv and beta doses from 10 μSv to 40 Sv. Main advantages are the flat energy response, the nondestructive readout and its small size.


**Fig. 5.12** DIS-1 personal dosimeter and reader (Photos courtesy of Mirion Technologies)

### *5.2.4 Chemical Dosimetry*

Chemical dosimeters are based on the change that some chemicals experiment as a result of induced oxidation–reduction reactions by the absorbed radiation. The response is expressed in terms of its sensitivity, known as radiation chemical yield —the old G value—defined by the number of chemical species produced or changed per joule of the energy absorbed in the solution.

The most common solutions are $Fe^{2+}$ (Fricke dosimeter) and $Ce^{4+}$ (cerium dosimeter), which can be used to measure absorbed doses up to 400 Gy and 1–50 kGy, respectively. Fricke dosimeter has found application as absolute dosimeter in the determination of absorbed dose to water at the primary standard level; also for the dosimetry of high activity gamma irradiators and 2–10 MeV electron accelerators.

The chlorobenzene-ethanol-trimethylpentane (CET) chemical dosimetry system is used in the accident and emergency personal dosimeter DL-M4 for a dose range from 1 to 10 Gy. Its design is a glass ampoule filled with the solution and inserted into a pen-shaped plastic holder. CET's response to neutrons has also been studied; it was found that the system could be used for the determination of total tissue dose due to nearly equivalent responses to gamma rays and neutrons.

Radiochromic detectors are solid-state materials that change their color as a result of a process of polymerization by absorbed radiation. The RADView™ PD radiation dosimeter badge is a credit card size radiation dosimeter incorporating a self-developing radiochromic film to provide visual measurement of absorbed

radiation dose for first responders in the event of an accident. The SIRAD® (Self-indicating Instant Radiation Alert Dosimeters) is other example of a radiochromic personal dosimeter for radiological incidents or accidents.

### 5.2.5 *Internal Dosimetry*

Internal exposure occurs when radionuclides have been inhaled, ingested, or otherwise taken into the body through wounds and intact skin. Measurements are used to calculate the intake of a radionuclide, which, when multiplied by the appropriate dose coefficient [3], leads to an estimate of committed effective dose.

Internal gamma emitters distributed throughout the body may be measured directly in vivo, which requires suitable arrangements of instruments. These body activity measurements are typically made using a number of large scintillation or semiconductor detectors, arranged above and below the subject, to increase the counting efficiency. Such systems, called whole body counters, are usually housed inside shielded rooms or booths to reduce background radiation and to allow even lower activities to be measured. Whole body counting might be carried out either using a static geometry or by scanning—moving the subject with respect to static detectors or moving detectors around a static subject. Systems have to be properly calibrated for the radionuclides of interest using phantoms made of tissue-equivalent materials to simulate specific measurement geometries. More complex phantoms which accurately represent the human body might be needed.

When radionuclides are concentrated in particular organs or tissues of the body, e.g., radioiodine—which is taken up by the thyroid—and inhaled radioactive particles—which are retained in the lungs—it is necessary to monitor specific sites. Localized monitoring is also used to assess the activity in contaminated wounds. $^{131}$I is normally monitored directly by measuring the activity in the thyroid using a simple NaI(Tl) detector. Compact arrays of three to six HPGe detectors are becoming standard for monitoring contamination in specific organs such as the lungs. Semiconductor cadmium telluride (CdTe) detectors operating at room temperatures are ideal for localized wound monitoring. They offer high sensitivity for detection of low-energy photons and have a small size (approximately 10 mm in diameter and 2 mm thick).

But direct methods are helpful for those radionuclides which emit photons of sufficient energy, and yield, to escape from the body and be measured by an external detector.

Monitoring from intakes of radionuclides can also be carried out indirectly by measuring the activity of biological samples such as excreta, breath, blood, or physical samples such as filters from personal or fixed air samplers, or surface smears. The number and timing of samples taken have to be carefully considered based on the biological and physical characteristics of the radionuclides to be

measured and other factors. Biokinetic models are needed to relate the activity level in an excreta sample to that in the body at the time the sample was taken, to relate the body content at the time the sample was taken to the original intake, and to calculate the committed effective dose from the estimated intake [13].

Liquid Scintillation Counting (LSC) is a routine laboratory method to monitor urine samples for tritium. The activity in fecal samples, nose blows, and nasal swabs can be determined using stationary or portable instruments. Fecal samples are typically taken to assess intakes of insoluble material. Breath is a significant route of excretion only for those few materials which are exhaled directly or metabolized to gases or volatile liquids.

Air concentration measurements, combined with assumptions about breathing rates and volumes and measured exposure times, can be used to estimate inhalation intakes. However, particle size influences the deposition of inhaled particulates in the respiratory tract; in many situations the airborne particle size distribution should be determined using cascade impactors or other methods.

Samples from area air monitoring provide an indication of the radionuclides and their relative concentrations in the work environment [14]. Personal, small, battery powered air pump samplers (PASs), which draw 2–4 L per minute of air through a filter are also used to routinely assess the probable intake of radionuclides and to select individuals for further assessments. PASs are worn with the air intake as close as possible to the nose and mouth of the worker. At the end of each working period, the filters are measured and the activity concentrations ($Bq.m^{-3}$) are calculated using the known air flow rate.

Gross count of total alpha activity can be made with a ZnS detector or a gas flow proportional counter. Alpha spectroscopy using semiconductor detectors can be used to quantify individual radionuclides after radiochemical separation. LSC is suitable to measure low-energy beta emitters. Gross measurements of high-energy beta emitters deposited on filters can be obtained using gas flow Geiger–Müller or proportional detectors.

Exposure to radon is of particular concern in underground mines, in buildings constructed with material containing significant levels of radium, in offices, factories, and other premises with elevated levels of uranium in the ground, and in buildings where large amounts of groundwater are processed [13]. Miners receive potentially significant lung doses from naturally occurring radon gases (especially $^{222}Rn$) and their solid decay products. Etched track detectors are widely used for the detection of radon and its decay products. A CR-39 dosimeter should be attached to the outside of a helmet or clothing. It might also be necessary to measure the gas/daughter equilibrium factor in the workplace to convert the measured gas concentrations to dose to workers.

Table 5.1 intends to generalize the instruments and detectors typically used for radiological protection and the type of radiation they measure. It also provides a short explanation about their function. Yet, when selecting instruments for any particular purpose it is important to take into account all the factors mentioned herein.

**Table 5.1** Summary of instruments used for measuring with radiation safety purposes

| | | |
|---|---|---|
| *Work place monitoring* | | |
| Contamination monitoring (skin, clothing, surfaces, airborne contamination sample meters, and alarm signals) | Proportional counters | Low-energy $\beta$ and $\alpha$ (gas flow), $\beta$-$\gamma$ counting |
| | Geiger–Müller counters | $\beta$-$\gamma$ counting, $\beta$ and $\alpha$ counting (thin windows) |
| | Scintillation counters | $\alpha$, $\beta$, $\gamma$, and neutron counting, $\gamma$ spectroscopy |
| | Liquid scintillation counting | Low-energy $\beta$ and $\alpha$ counting, $\beta$ radiation, and Auger electrons |
| | Ionization chambers | *X*-ray and $\beta$-$\gamma$ counting |
| | Semiconductor detectors | $\alpha$ counting, $\alpha$ spectroscopy, radon in air, $\gamma$ spectroscopy |
| Dose rate monitoring (dose ratemeters, integrating dose ratemeters for external dose and shielding assessment, and alarm signals) | Proportional counters filled with $BF_3$ or $^3He$ | Neutron |
| | Proportional counters | *X*-rays spectroscopy |
| | Geiger–Müller counters | $\gamma$ survey meters |
| | Scintillation counters | $\gamma$ and neutron survey meters |
| | Ionization chambers | $\gamma$, $\beta$, low-energy *X*-rays and neutron dose measurements, survey meters |
| | Silicon solid-state detectors | $\alpha$, $\beta$, *X*-rays survey meters, $\alpha$ spectroscopy |
| | Solid-state nuclear track detector | Neutron, $\alpha$ |
| *Individual monitoring* | | |
| External dose | Film badge dosimeters | $\gamma$, *X*-rays, $\beta$, thermal neutrons |
| | Thermoluminescent (TLD) dosimeters | $\gamma$, *X*-rays, $\beta$, thermal neutrons |
| | Radiophotoluminescent glass dosimeters | $\gamma$, low-energy photons |
| | Solid-state nuclear track detectors | Neutrons (fast, intermediate, and thermal) |
| | Silicon solid-state detector | $\gamma$, $\beta$, neutrons (direct reading) |
| | Optically stimulated luminescence dosimeters | $\gamma$, *X*-rays, $\beta$ (neutron with CR-39 incorporated) |
| | Electronic personal dosimeters | $\gamma$, *X*-rays, $\beta$ (direct reading) |
| | Direct ion storage dosimeters | $\gamma$, *X*-rays, $\beta$ (some neutrons) (direct reading) |

(continued)

**Table 5.1** (continued)

| *Individual monitoring* | | |
|---|---|---|
| Activity due to intake of radionuclides | Scintillation counters | Whole body counting, $\gamma$ spectroscopy |
| | Liquid scintillation counting | Activity in biological samples |
| | Ionization chambers | Organ counting, activity in biological samples, contaminated wounds |
| | Semiconductor detectors | Whole body counting, $\gamma$ spectroscopy, $\alpha$ spectroscopy, contaminated wounds |
| | Proportional counters | Activity in biological samples, contaminated wounds |
| | Etched track detectors | Radon detection |

All instruments must have detection efficiencies ranging from zero to 30 % (at best) for different radionuclides. Measurements must be made using a calibrated instrument with the best available, predetermined detection efficiency

## References


1. International atomic energy (2004) Workplace monitoring for radiation and contamination, practical radiation technical manual 1 (Rev. 1). IAEA, Vienna
2. Doerfel H, et al. (2008) IRPA12. RC-6. Internal dosimetry. The science and art of internal dose assessment. www.irpa.net/images/stories/irpa12/refresher-courses/RC%206%20Internal%20Dosimetry%20-%20The%20Science%20and%20Art%20of%20Internal%20Dose%20Assessment.pdf. Accessed 15 April 2015
3. ICRP publication 119 (2012) Compendium of dose coefficients based on ICRP publication 60. Ann ICRP 41(1)
4. International atomic energy, individual monitoring (2004) Practical radiation technical manual 2 (Rev. 1). IAEA, Vienna
5. Huang DY, Hsu S-M (2008) Radio-photoluminescence glass dosimeter (RPLGD). Med Phys 35:5558. http://cdn.intechopen.com/pdfs-wm/23935.pdf. Accessed 21 Nov 2008
6. Juto N (2002) The large scale personal monitoring service using the latest personal monitor GLASS BADGE. In: First Asian and oceanic congress for radiation protection (AOCRP-1), Seoul
7. International Atomic Energy (2005) Radiation oncology physics: a handbook for teachers and students, STI/PUB/1196. IAEA, Vienna
8. Frame PW (2005) A history of radiation detection instrumentation. Health Phys 88(6):613–637
9. Landauer®, Luxel® + Dosimeter for X, Gamma, Beta and Neutron Radiation (2005). Landauer, Inc, Glenwood
10. Canberra data sheet (2008) DOSICARD™—DOSIMAN electronic personal dosimeters in credit card size. Canberra
11. Mirion Technologies Health Physics Division (2014) DIS-1 Dosimeter user's guide, Document Number 2096 6058, Version 1.04. Mirion Technologies

12. Wernli C, Fiechtner A, Kahilainen J. Neutron dosimetry with ion chamber-based DIS system. RADOS Technology Oy. http://www.irpa.net/irpa10/cdrom/00090.pdf
13. International atomic energy, international labour office (1999) Safety guide No. RS-G-1.2. Assessment of occupational exposure due to intakes of radionuclides. IAEA, Vienna
14. International Atomic Energy (2004) Methods for assessing occupational radiation doses due to intakes of radionuclides, Safety report series no. 37. IAEA, Vienna

# Chapter 6
# Dose Assessment

Dose[1] calculation could be, and in fact is, a very complex task which, in many cases, needs simulation using numerical integration. A dose reconstruction after an emergency situation, or a careful evaluation of the absorbed dose in any exposure situation, requires accurate estimates which depend upon the body modeling, the radiation environment modeling, and the Monte Carlo treatment.

Monte Carlo simulation is a class of computational algorithms that rely on repeated random sampling to compute their results. These methods have proven to be very efficient in solving differential equations of radiation fields and energy transport. They can simulate the interaction of radiation with matter on the basis of the information from nuclear data with practically no restrictions on the geometry of considered systems.

In the past two decades, a number of Monte Carlo computer programs have been developed for high-energy physics research and are in use in the field of radiological imaging and radiation dosimetry. The Monte Carlo Universal (MCU) [1], a project started at Kurchatov Institute in 1982, has made available various computer codes for simulation of particle transport (neutrons, photons, electrons) in three-dimensional systems. The Monte Carlo N-Particle Transport Code (MCNP) [2], a software package developed and owned by Los Alamos National Laboratory is used primarily for the simulation of nuclear processes such as fission, but has the capability to simulate particle interactions involving neutrons, photons, and electrons. The MCNPX [3], also developed by Los Alamos National Laboratory, is capable of simulating particle interactions of 34 different types of particles at all energies, including those simulated by MCNP. Both codes can be used to determine doses from sources.

[1]Hereinafter, dose will mean "absorbed dose" if not otherwise stated.



H. Domenech, *Radiation Safety*, DOI 10.1007/978-3-319-42671-6_6

The use of dynamic anthropomorphic models (phantoms) in Monte Carlo simulations made rapid headway as well. From 2007 to 2011, the Consortium of Computational Human Phantoms (CCHP) [4] was recognized as an international initiative for promoting collaborative research in computational modeling of the human body. Computational phantoms are highly sophisticated tools that have become possible as a result of the increasing availability of computer power and advanced medical imaging technologies [5, 6].

For example, the VMC model (Visual Monte Carlo Program) with 20 years of development, is a computer Monte Carlo program, Windows based, that simulates the irradiation of the human body by various radiation sources, including point, ground, cloud, or internal sources using a voxel[2] phantom to solve radiation protection problems [7, 8]. VMC is developed into two main softwares—VMC in vivo for simulation of a whole-body counter laboratory, and VMC dose calculation, for dose calculations due to exposure to radionuclides or *X*-rays.

Apart from the fact that accurate computing techniques are indeed necessary in radiation dosimetry and for dose reconstruction in emergencies or accidents; the assessment of dose using prudently straightforward approximations is still a useful tool in radiation safety to prior evaluate the risk from a certain exposure situation, to determine the measurement equipment or method to be used, and for reporting purposes, etc.

## 6.1 Dose from an External Point Gamma Source

In routine assessments of external exposure, the dose from a gamma source of a single radionuclide could be calculated assuming the source is an isotropic point source, i.e., a small volume source uniformly emitting in all directions ($4\pi$) with negligible self-attenuation in an infinite homogeneous medium. This approximation is justified if the distance from the center of the source to the point of interest is at least three times the largest source dimension.

We should also assume that the source acts to the point of interest at a distance $r$ and that the medium is air. The effective dose can then be determined according to the inverse square law, which states that the intensity of the radiation decreases in proportion to the inverse of the distance from the source squared, thus

$$\mathrm{E} = \frac{\mathrm{A}_0 \cdot \Gamma \cdot \left(\mathrm{t} - \mathrm{e}^{-\lambda \mathrm{t}}\right)}{\lambda \cdot \mathrm{r}^2}, \text{unit Gy},$$

[2]Voxel is a blend of the words volumetric and pixel. Simplifying it is as a way to represent volumetric objects as 3D bitmaps instead of vectors. A voxel represents the elementary tissue volume that corresponds to a pixel in an image.

where

$\Gamma$ = kerma constant (former gamma constant for non-SI units) in $Gy.m^2.Bq^{-1}.s^{-1}$; a conversion factor that relates the unit of activity for a specific radionuclide to kerma rate at 1 m in air

$A_0$ = radionuclide initial activity, Bq

$\lambda$ = decay constant for the radionuclide, $s^{-1}$

t = length of exposure, s

r = distance from the source to the point of interest, m

If the product of $\lambda t$ is negligible and shielding and air attenuation is disregarded, the following equation may be used:

$$E = \frac{\Gamma \cdot A_0 \cdot t}{r^2}$$

Given that the degree of production of charged particles by gamma ray interaction in air depends on the photons' energies, kerma constant might be determined by

$$\Gamma_\delta = \frac{1.602 \cdot 10^{-13}}{4 \cdot \Pi} \cdot \sum_j n_j \cdot hv_j \cdot \left(\mu_{\Delta,m}\right)_j, \text{ unit } Gy.m^2.Bq^{-1}.s^{-1},$$

where

$1.602.10^{-13}$ = constant to convert MeV to joules (J);

$n_j$ = number of j photons emitted by each disintegration; $Bq^{-1}.s^{-1}$

$hv_j$ = j photon energy, MeV. The subscript $\delta$ implies that only photons with energy > $\delta$ are considered for calculation;

$(\mu_{\Delta,m})_j$ = mass energy-transfer coefficient of each j photon, $m^2.kg^{-1}$

Photon mass attenuation coefficients and mass energy-absorption coefficients can be found online at the Physics Laboratory of the National Institute of Standards and Technology (NIST) website [9]. Also, mass attenuation coefficients and mass energy-absorption coefficients for all elements from Z = 1–92 [10].

Gamma constants—dose equivalent rate at 1 m from a point source—for approximate 500 radionuclides important to dosimetry and radiological assessment were published in ORNL RSIC-45/R1 [11] in 1982. The same data was later published in 1992 in "The Health Physics and Radiological Health Handbook" [12]. All published data are based on calculations assuming a point source in vacuum, but use different cutoff energies. That is why they may not agree. For example, Ninkovic [13] and Wasserman [14] took into account gamma rays and characteristic *X*-ray photons with energies >20 keV; the data published in Kaye and Laby online [15] use a cutoff of 50 keV, and Unger [11] included only gamma rays of energy >10 keV.

Kerma constant can also be obtained from the Radiological Toolbox software developed by ORNL [16]. This is essentially an electronic handbook containing databases needed in radiation protection, shielding, and dosimetry calculations, and

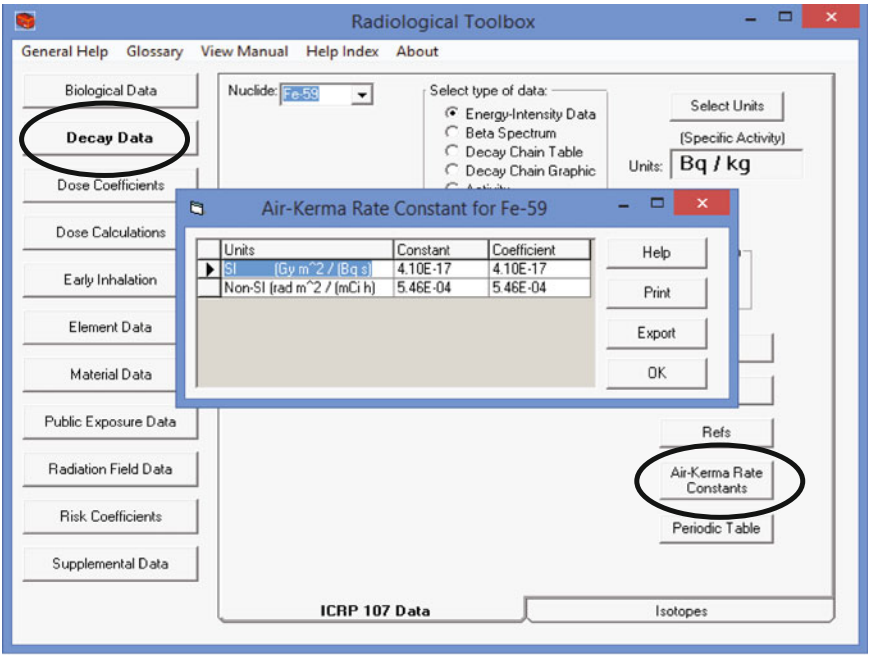


**Fig. 6.1** Screenshot from Radiological Toolbox ["Courtesy of Oak Ridge National Laboratory, U.S. Dept. of Energy"]

can display the air kerma rate constant for specific radionuclides using updated decay data. A screen caption from version 2013 is shown in Fig. 6.1.

## 6.2 Dose from External Beta Sources

Most beta particles do not normally penetrate beyond the epidermal layer of the skin, but when sufficiently intense, they are capable of depositing their energy over a few millimeter depth of tissue, thus, causing burns and/or tissue damage. Equivalent doses to skin resulting from clothing contamination or from dermal contamination by direct contact with contaminated objects could be locally high.

The conversion factors in Table 6.1 below can be used to make a rapid assessment of the beta dose to skin from a suspected contamination or to previously assess a planned operation using the following expression. The contamination is supposed to be uniformly and thinly spread over the skin.

$$H_{T(skin)} = \frac{C_{skin} \cdot CF_{beta-skin} \cdot t}{SF_{beta-skin}},$$

**Table 6.1** Conversion factors for beta dose to skin ["Downloaded from the RADAR website [17], a courtesy of the Radiation Dose Assessment Resource (RADAR) Team".]

| Radionuclide | $CF_{Beta\text{-}skin}$ ($\mu Gy.h^{-1}.Bq^{-1}.cm^2$) | Radionuclide | $CF_{Beta\text{-}skin}$ ($\mu Gy.h^{-1}.Bq^{-1}.cm^2$) | Radionuclide | $CF_{Beta\text{-}skin}$ ($\mu Gy.h^{-1}.Bq^{-1}.cm^2$) |
|---|---|---|---|---|---|
| $^{3}H$ | 0 | $^{68}Ga$ | 1.8 | $^{131}I$ | 1.6 |
| $^{14}C$ | 0.32 | $^{76}As$ | 2.1 | $^{131}Cs$ | 0.01 |
| $^{18}F$ | 1.9 | $^{75}Se$ | 0.14 | $^{134}Cs$ | 1.4 |
| $^{22}Na$ | 1.7 | $^{77}Br$ | 0.01 | $^{137}Cs$ | 1.6 |
| $^{24}Na$ | 2.2 | $^{82}Br$ | 1.5 | $^{133}Ba$ | 0.13 |
| $^{26}Al$ | 1.8 | $^{87}Rb$ | 1.9 | $^{140}Ba/^{140}La$ | 3.8 |
| $^{32}P$ | 1.9 | $^{85}Sr$ | 0.06 | $^{139}Ce$ | 0.49 |
| $^{33}P$ | 0.86 | $^{89}Sr$ | 1.8 | $^{141}Ce$ | 1.8 |
| $^{35}S$ | 0.35 | $^{90}Sr/^{90}Y$ | 3.5 | $^{143}Ce$ | 2.0 |
| $^{36}Cl$ | 1.8 | $^{90}Y$ | 2.0 | $^{152}Eu$ | 0.92 |
| $^{40}K$ | 1.5 | $^{95}Zr/^{95}Nb$ | 1.6 | $^{154}Eu$ | 2.1 |
| $^{45}Ca$ | 0.84 | $^{99}Mo/^{99m}Tc$ | 1.9 | $^{186}Re$ | 1.8 |
| $^{51}Cr$ | 0.015 | $^{99m}Tc$ | 0.25 | $^{188}Re$ | 2.3 |
| $^{56}Mn$ | 2.4 | $^{99}Tc$ | 1.2 | $^{192}Ir$ | 1.9 |
| $^{59}Fe$ | 0.97 | $^{110m}Ag$ | 0.68 | $^{198}Au$ | 1.7 |
| $^{56}Co$ | 0.55 | $^{111}Ag$ | 1.8 | $^{197}Hg$ | 0.092 |
| $^{57}Co$ | 0.12 | $^{111}In$ | 0.38 | $^{203}Hg$ | 0.89 |
| $^{58}Co$ | 0.30 | $^{113m}In$ | 0.73 | $^{201}Tl$ | 0.27 |
| $^{60}Co$ | 0.78 | $^{115m}In$ | 1.3 | $^{204}Tl$ | 1.6 |
| $^{65}Ni$ | 2.2 | $^{132}Te$ | 0.78 | $^{210}Pb$ | 0.0084 |
| $^{64}Cu$ | 1.0 | $^{123}I$ | 0.38 | $^{235}U$ | 0.18 |
| $^{67}Cu$ | 1.3 | $^{124}I$ | 0.52 | $^{241}Am$ | 0.019 |
| $^{65}Zn$ | 0.076 | $^{125}I$ | 0.021 | | |

where

$H_{T(skin)}$ = equivalent dose to the skin at 70 μm in depth, mGy

$C_{skin}$ = average surface concentration of radionuclide on skin or clothing, $Bq.cm^{-2}$

$CF_{Beta\text{-}skin}$ = conversion factor: skin beta dose rate, $\mu Gy.\ h^{-1}.Bq^{-1}.cm^2$

$SF_{Beta\text{-}skin}$ = shielding factor due to clothing; representative values of shielding factors are approximately 3–5 for light clothing and 1000 for heavy clothing;

t = length of exposure, h

The equivalent dose is calculated for each radionuclide and then summed to obtain the total equivalent dose.

Beta particles can also produce bremsstrahlung radiation in their interaction with the source material or shielding. Bremsstrahlung is particularly important in the case of high energy (>1 MeV) beta emitters, e.g., $^{32}P$ with maximum energy of 1.7 MeV, and cannot be disregarded.

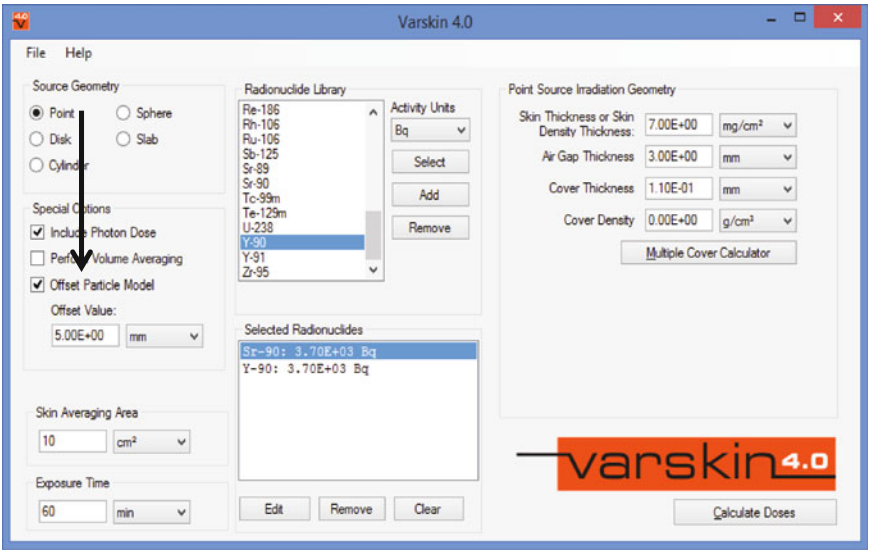


**Fig. 6.2** Screenshot from VARSKIN 4

The intensity of bremsstrahlung radiation is proportional to the energy of the beta particles and the atomic number of the material through which the betas are passing. The fraction of beta particle or monoenergetic electron energy converted to photons when absorbed by a material with atomic number Z, is

$$f_\beta = 3.5 \cdot 10^{-4} Z E_\beta$$
$$f_e = 10^{-4} Z E_e,$$

where $E_\beta$ and $E_e$ are the maximum beta and electron energy, respectively, in MeV.

Effective bremsstrahlung photon energy ($hv$) can also be easily obtained from maximum beta energy $E_{max}$:

$$hv = \frac{1}{2} E_{\max}, \quad \text{for } E_{max} \leq 10\,\text{MeV}$$
$$hv = \frac{1}{3} E_{\max}, \quad \text{for } 10\,\text{MeV} < E_{max} < 30\,\text{MeV}$$

VARSKIN is a Monte Carlo-based computer code system designed to calculate the dose to skin from radioactive contamination of skin or protective clothing [18] available from the U.S. Nuclear Regulatory Commission.[3] As seen from the screenshot of VARSKIN version 4.0 in Fig. 6.2 ["Courtesy of Oak Ridge National

[3]Pursuant to Title 17 Sect. 105 of the United States, Code VARSKIN is not subject to copyright protection and is in the public domain.

Laboratory, U.S. Dept. of Energy"], five different predefined source configurations are available—point, disk, cylinder, sphere, and slab.

The program offers the option to calculate doses from multiple sources selecting various radionuclides from the list. Also, it calculates the dose from multiple hot particles. Hot particles are small, discrete, highly radioactive particles capable of causing extremely high doses to a localized area in a short time period. These particles differ radically from uniform skin contamination in that the particles have a thickness associated with them. For this calculation, the option "Offset Particle Model" appears when the point source geometry is selected. The offset value is the lateral distance between the point source and the center of the dose area.

## 6.3 Dose from Neutrons

Fluence to effective dose conversion coefficients as a function of incident neutron energy on Table 6.2 are reprinted from Publication 116 [19] with permission of the ICRP and, according to this publication, were calculated assuming a whole-body irradiation of the ICRP/ICRU Reference Adult Male and Reference Adult Female [20] phantoms, placed in a vacuum, coupled to various Monte Carlo radiation transport codes (MCNPX, PHITS, FLUKA), by broad unidirectional beams assumed to represent occupational exposures.

AP, PA, and LAT[4] geometries are considered to approximate radiation fields produced by single sources at large distances and particular body orientations, and thus they approximate real occupational exposure geometries. The ISO geometry approximates the radiation field to which a body is subjected, if suspended in a large cloud of radioactive gas or placed in a highly scattered radiation field, e.g., irradiation by naturally occurring radionuclides in homes or the environment, or by atmospheric releases of radionuclides into the environment [19].

Monte Carlo calculations take account of the production of secondary particles and photons by neutron-induced reactions or elastic scattering. Every secondary charged particle is tracked and the resulting energy deposition in each organ is summed to compute organ absorbed doses; fluence to effective dose conversion coefficients are then derived from the organ dose conversion coefficients. ICRP/ICRU reference phantoms are shown in Fig. 6.3.

Coefficients on Table 6.3 are reprinted form Publication 119 [21] with the permission of ICRP. These coefficients, published by ICRP in 1996 [22], and that also appear in the International Basic Standards [23], were calculated for operational quantities using stylized phantoms and the radiation weighting factors for neutrons from 1990's Commission recommendations [24].

[4]AP = anteroposterior; PA = posteroanterior; LLAT = left lateral, and ISO = isotropic.

**Table 6.2** Effective dose per unit neutron fluence from ICRP Publication 116

| Energy (MeV) | E/Φ (pSv.cm$^2$) | | | | Energy (MeV) | E/Φ (pSv.cm$^2$) | | | |
|---|---|---|---|---|---|---|---|---|---|
| | AP | PA | LLAT | ISO | | AP | PA | LLAT | ISO |
| 1.0E-09 | 3.09 | 1.85 | 1.04 | 1.29 | 3.0 | 500 | 398 | 310 | 327 |
| 1.0E-08 | 3.55 | 2.11 | 1.15 | 1.56 | 4.0 | 500 | 404 | 316 | 332 |
| 2.5E-08 | 4.00 | 2.44 | 1.32 | 1.76 | 5.0 | 499 | 412 | 325 | 339 |
| 1.0E-07 | 5.20 | 3.25 | 1.70 | 2.26 | 6.0 | 495 | 417 | 333 | 344 |
| 2.0E-07 | 5.87 | 3.72 | 1.94 | 2.54 | 7.0 | 493 | 419 | 336 | 346 |
| 5.0E-07 | 6.59 | 4.33 | 2.21 | 2.92 | 8.0 | 490 | 420 | 338 | 347 |
| 1.0E-06 | 7.03 | 4.73 | 2.40 | 3.15 | 9.0 | 484 | 422 | 343 | 350 |
| 2.0E-06 | 7.39 | 5.02 | 2.52 | 3.32 | 10.0 | 477 | 423 | 347 | 352 |
| 5.0E-06 | 7.71 | 5.30 | 2.64 | 3.47 | 12.0 | 330 | 167 | 130 | 153 |
| 1.0E-05 | 7.82 | 5.44 | 2.65 | 3.52 | 14.0 | 365 | 195 | 150 | 174 |
| 2.0E-05 | 7.84 | 5.51 | 2.68 | 3.54 | 15.0 | 407 | 235 | 179 | 203 |
| 5.0E-05 | 7.82 | 5.55 | 2.66 | 3.55 | 16.0 | 458 | 292 | 221 | 244 |
| 1.0E-04 | 7.79 | 5.57 | 2.65 | 3.54 | 18.0 | 483 | 330 | 249 | 271 |
| 2.0E-04 | 7.73 | 5.59 | 2.66 | 3.52 | 20.0 | 494 | 354 | 269 | 290 |
| 5.0E-04 | 7.54 | 5.60 | 2.62 | 3.47 | 30.0 | 453 | 422 | 360 | 358 |
| 0.001 | 7.54 | 5.60 | 2.61 | 3.46 | 50.0 | 433 | 428 | 380 | 371 |
| 0.002 | 7.61 | 5.62 | 2.60 | 3.48 | 75.0 | 420 | 439 | 399 | 387 |
| 0.005 | 7.97 | 5.95 | 2.74 | 3.66 | 100 | 402 | 444 | 409 | 397 |
| 0.01 | 9.11 | 6.81 | 3.13 | 4.19 | 130 | 382 | 446 | 416 | 407 |
| 0.02 | 12.2 | 8.93 | 4.21 | 5.61 | 150 | 373 | 446 | 420 | 412 |
| 0.03 | 15.7 | 11.2 | 5.40 | 7.18 | 180 | 363 | 447 | 425 | 421 |
| 0.05 | 23.0 | 15.7 | 7.91 | 10.4 | 200 | 359 | 448 | 427 | 426 |
| 0.07 | 30.6 | 20.0 | 10.5 | 13.7 | 300 | 363 | 464 | 441 | 455 |

(continued)

**Table 6.2** (continued)

| Energy (MeV) | E/Φ (pSv.cm$^2$) | | | | Energy (MeV) | E/Φ (pSv.cm$^2$) | | | |
|---|---|---|---|---|---|---|---|---|---|
| | AP | PA | LLAT | ISO | | AP | PA | LLAT | ISO |
| 0.1 | 41.9 | 25.9 | 14.4 | 18.6 | 400 | 389 | 496 | 472 | 488 |
| 0.15 | 330 | 167 | 130 | 153 | 500 | 422 | 533 | 510 | 521 |
| 0.2 | 365 | 195 | 150 | 174 | 600 | 457 | 569 | 547 | 553 |
| 0.3 | 407 | 235 | 179 | 203 | 700 | 486 | 599 | 579 | 580 |
| 0.5 | 458 | 292 | 221 | 244 | 800 | 508 | 623 | 603 | 604 |
| 0.7 | 483 | 330 | 249 | 271 | 900 | 524 | 640 | 621 | 624 |
| 0.9 | 494 | 354 | 269 | 290 | 1000 | 537 | 654 | 635 | 642 |
| 1.0 | 498 | 371 | 284 | 303 | 2000 | 612 | 740 | 730 | 767 |
| 1.2 | 499 | 383 | 295 | 313 | 5000 | 716 | 924 | 963 | 1.01E + 3 |
| 2.0 | 499 | 392 | 303 | 321 | 10,000 | 933 | 1.17E + 3 | 1.23E + 3 | 1.32E + 3 |

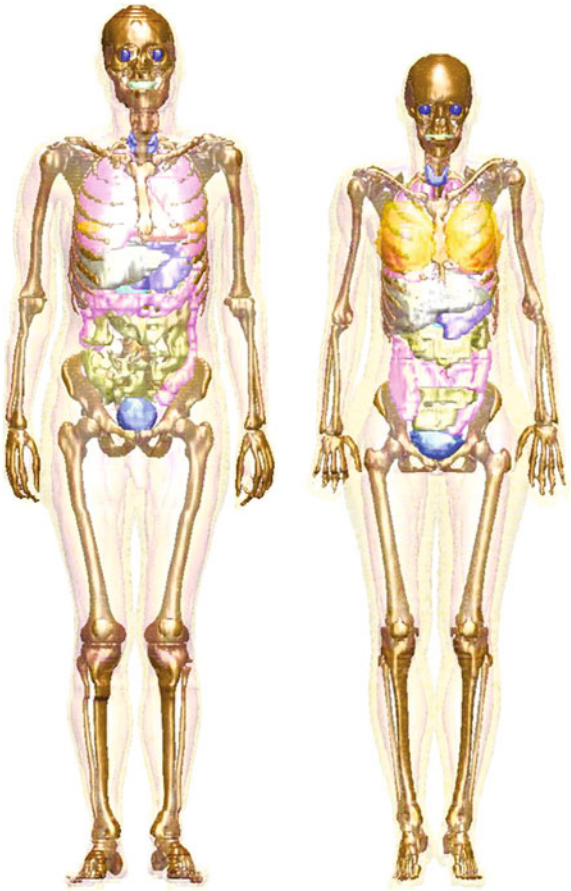

**Fig. 6.3** ICRP/ICRU Reference Adult Male and Reference Adult Female phantoms. Breast, bones, colon, eyes, lungs, liver, pancreas, small intestine, stomach, teeth, thyroid, and urinary bladder are identified by different surface colors. Muscle and adipose tissue are displayed as transparent ["Courtesy of Oak Ridge National Laboratory, U.S. Dept. of Energy"]

For a radioisotope neutron source, assuming it is a point source, the neutron fluence for the given energy, $E_n$, at a distance $r$, $\Phi(E_n)$, can be obtained by

$$\Phi(E_n) = \frac{A_e(E_n) \cdot t}{4 \cdot \Pi \cdot r^2}, \quad \mathrm{m}^{-2},$$

where

$A_e(E_n)$ = number of neutrons with energy $E_n$ emitted by the source, $s^{-1}$

t = length of exposure, s

With this fluence, and the effective dose conversion coefficient for energy $E_n$, it is possible to roughly estimate the neutron effective dose, but bearing in mind that all neutrons are treated as if they had the average neutron energy $E_n$. Computing the actual spectrum of neutrons also requires Monte Carlo method. This method allows to mathematically constructing a detailed three-dimensional geometrical model to simulate interactions based on the cross section for the interaction with the specific material at that neutron energy, and following the consequences.

**Table 6.3** Effective dose per unit neutron fluence from ICRP Publication 119

| Energy (MeV) | E/Φ (pSv.cm$^2$) | | | | Energy (MeV) | E/Φ (pSv.cm$^2$) | | | |
|---|---|---|---|---|---|---|---|---|---|
| | AP | PA | LLAT | ISO | | AP | PA | LLAT | ISO |
| 1.0E-09 | 5.24 | 3.52 | 1.36 | 2.40 | 0.15 | 80.2 | 52.2 | 21.2 | 35.2 |
| 1.0E-08 | 6.55 | 4.39 | 1.7 | 2.89 | 0.2 | 99.0 | 61.5 | 25.6 | 42.4 |
| 2.5E-08 | 7.60 | 5.16 | 1.99 | 3.30 | 0.3 | 133 | 77.1 | 33.4 | 54.7 |
| 1.0E-07 | 9.95 | 6.77 | 2.58 | 4.13 | 0.5 | 188 | 103 | 46.8 | 75.0 |
| 2.0E-07 | 11.2 | 7.63 | 2.92 | 4.59 | 0.7 | 231 | 124 | 58.3 | 92.8 |
| 5.0E-07 | 12.8 | 8.76 | 3.35 | 5.20 | 0.9 | 267 | 144 | 69.1 | 108 |
| 1.0E-06 | 13.8 | 9.55 | 3.67 | 5.63 | 1.0 | 282 | 154 | 74.5 | 116 |
| 2.0E-06 | 14.5 | 10.2 | 3.89 | 5.96 | 1.2 | 310 | 175 | 85.8 | 130 |
| 5.0E-06 | 15.0 | 10.7 | 4.08 | 6.28 | 2.0 | 383 | 247 | 129 | 178 |
| 1.0E-05 | 15.1 | 11.0 | 4.16 | 6.44 | 3.0 | 432 | 308 | 171 | 220 |
| 2.0E-05 | 15.1 | 11.1 | 4.20 | 6.51 | 4.0 | 458 | 345 | 198 | 250 |
| 5.0E-05 | 14.8 | 11.1 | 4.19 | 6.51 | 5.0 | 474 | 366 | 217 | 272 |
| 1.0E-04 | 14.6 | 11.0 | 4.15 | 6.45 | 6.0 | 483 | 380 | 232 | 282 |
| 2.0E-04 | 14.4 | 10.9 | 4.10 | 6.32 | 7.0 | 490 | 391 | 244 | 290 |
| 5.0E-04 | 14.2 | 10.7 | 4.03 | 6.14 | 8.0 | 494 | 399 | 253 | 297 |
| 0.001 | 14.2 | 10.7 | 4.00 | 6.04 | 9.0 | 497 | 406 | 261 | 303 |
| 0.002 | 14.4 | 10.8 | 4.00 | 6.05 | 10.0 | 499 | 412 | 268 | 309 |
| 0.005 | 15.7 | 11.6 | 4.29 | 6.52 | 12.0 | 499 | 422 | 278 | 322 |
| 0.01 | 18.3 | 13.5 | 5.02 | 7.70 | 14.0 | 496 | 429 | 286 | 333 |
| 0.02 | 23.8 | 17.3 | 6.48 | 10.2 | 15.0 | 494 | 431 | 290 | 338 |
| 0.03 | 29.0 | 21.0 | 7.93 | 12.7 | 16.0 | 491 | 433 | 293 | 342 |
| 0.05 | 38.5 | 27.6 | 10.6 | 17.3 | 18.0 | 486 | 435 | 299 | 345 |
| 0.07 | 47.2 | 33.5 | 13.1 | 21.5 | 20.0 | 480 | 436 | 305 | 343 |
| 0.1 | 59.8 | 41.3 | 16.4 | 27.2 | | | | | |

## 6.4 Dose from *X*-Rays

The production of *X*-rays involves the bombardment of a target with energetic electrons. These electrons undergo a complex sequence of collisions and scattering processes during the slowing down process, which results in the production of bremsstrahlung and characteristic radiation. Thus, radiation output from a *X*-ray tube depends on the anode material, the atomic number of the element it is made of, the applied kilovoltage to the tube, that defines the kinetic energy of the electrons bombarding the anode, and the amount and type of added filtration.

Also, the radiation emitted by an *X*-ray tube is heterogeneous, that is, it contains *X*-rays of a number of wavelengths, in the form of a continuous spectrum with superimposed intensity spikes at energies that are characteristic of the metal used to make the target. It also depends on the inherent filtration in the *X*-ray tube window (glass, aluminum, beryllium, etc.).

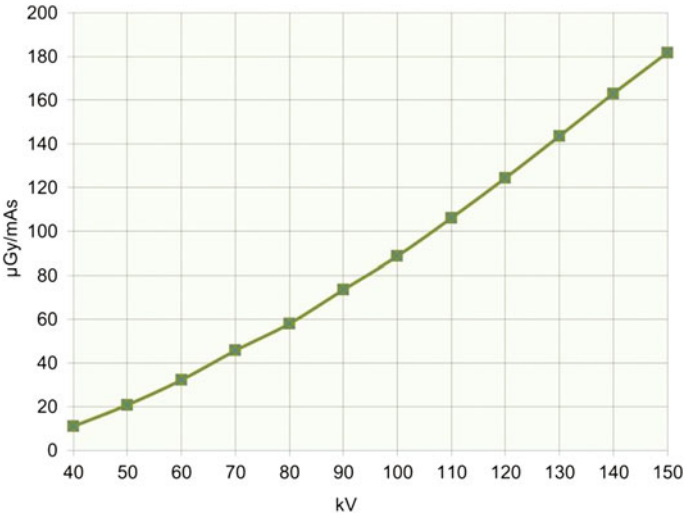


**Fig. 6.4** Specific yield of an *X*-rays tube

The kerma rate in air (Gy/min) at a certain distance (typically 1 m) from the focal spot can be estimated with reasonable accuracy for a given voltage, filament current, and useful beam filtration, when the specific yield of the *X*-ray tube (mGy/mA·s) is known. The specific yield depends on tube age, construction, etc., and can be measured to obtain the yield curve normalized to 1 m. A typical example is shown in Fig. 6.4.

The tube output and the air kerma can be simulated using Monte Carlo methods. A general purpose Monte Carlo code system for the simulation of coupled electron–photon transport in arbitrary materials like PENELOPE is a good example [25].

The conversion coefficients for effective dose per air kerma for photons of energies up to 10 MeV are tabulated in ICRP Publication 116 [19] in units of Sv/Gy.

## 6.5 Dose Due to Intake of Radionuclides

While the main route of intake in occupational exposure is by inhalation, a fraction of any material deposited in the respiratory system may be transferred to the throat and swallowed, giving the opportunity for absorption in the gastrointestinal tract. Intakes by direct ingestion may also occur—for example, when a worker touches his or her mouth with contaminated hands—and, for some radionuclides like $^{3}H$, absorption through the intact skin. Damage to the skin by cuts or other wounds can also result in intakes of radionuclides.

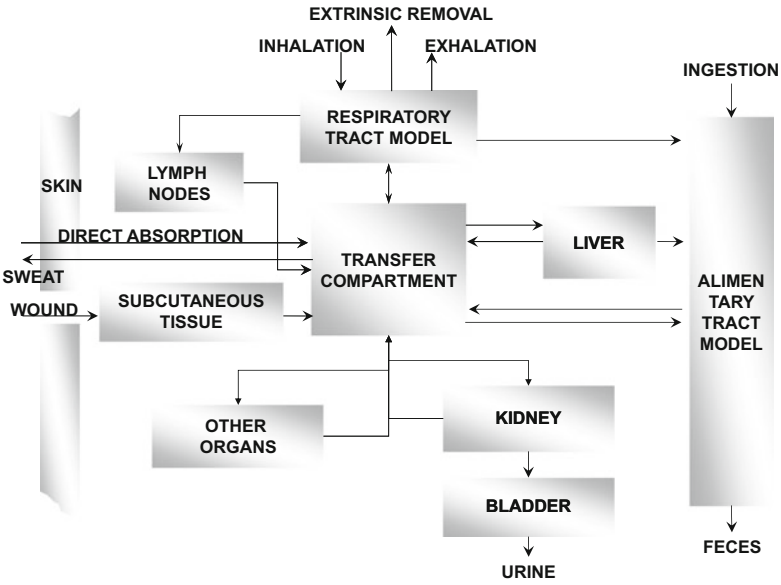


**Fig. 6.5** Routes of intake, transfers, and excretion based on ICRP Publication 54 ["Reproduced with permission by the IAEA."]

Routes of intake, transfers, and excretion are represented in Fig. 6.5 [26]. The fraction of the radionuclide entering into the body is called "the intake". The fraction absorbed into the blood, and hence entering the body fluids, after an intake has occurred is called "the uptake". Transfers govern the radionuclide distribution into the body and its route and rate of elimination.

Intake, uptake, internal transfer, and excretion of radionuclides can be described by means of various compartmental models developed by ICRP. The human respiratory tract model (HRTM) [27] describes the behavior of radionuclides inhaled by workers. The behavior of radionuclides ingested by workers is described by the Human Alimentary Tract Model (HATM) [28], a model based on four gastrointestinal tract compartments representing the stomach, the small intestine, the upper large intestine, and the lower large intestine. Specific biokinetic models for systemic radionuclides have also been developed to describe the time-dependent distribution and retention of selected radionuclides in the body after it reaches the systemic circulation, and its excretion from the body [29–31].

After an uptake, the fraction of the radionuclide remaining in the body or being excreted from the body depends on its effective half-life and is a function of the time period since the intake.

$$T_e = \frac{T_{1/2} \cdot T_b}{T_{1/2} + T_b},$$

where $T_{1/2}$ is the radioactive half-life and $T_b$, the biological half-life. The effective half-life is the time taken for the amount of a radionuclide deposited in a living organism to be reduced by 50 % as a result of the combined action of radioactive decay and biological elimination.

The effective half-life is associated with the effective decay constant ($\lambda_e$), which is equal to the sum of the biological decay constant and the physical decay constant.

$$\lambda_e = \lambda_{1/2} + \lambda_b$$

In other words, the time a radionuclide is in a given organ or tissue depends on both its physical decay constant—radioactive decay—and the biological removal mechanism related to its physical–chemical behavior within the body.

Dose assessment is based on the measurement of the intake of a radionuclide either by direct or indirect methods. Biokinetic models are used to interpret the measurements and the effective dose is calculated from the intake using reference dose coefficients. Alternatively, the committed effective dose can be calculated directly from the measurements using functions that relate them to the time of the intake. These functions are tabulated as "dose per unit content".

Intake, uptake, and excretion by inhalation and ingestion are described by the respiratory tract (HRTM) [27] and the alimentary tract (HATM) [28] models, respectively. Biokinetic models can be used both prospective—in assessing radiation dose and hence a potential risk—and retrospective—in assessing the amount of radioactivity an individual has inhaled or ingested.

The respiratory tract model is represented by five regions: (1) extrathoracic airways, divided into anterior nose and posterior nasal passages, pharynx, and larynx; (2) bronchial; (3) bronchiolar; (4) alveolar–interstitial; and (5) gas exchange region. Lymphatic tissue is associated with the extrathoracic and thoracic airways, respectively. Route of the radionuclides is described in terms of mathematical equations for each region of the respiratory tract, with account taken of both inhalation and exhalation.

The deposition of particulate material in the upper respiratory tract is governed by particle size, breathing parameters and/or work load, and is not considered depending on the chemical form. Clearance in this region is treated as two competing processes: particle transport—by mucociliary clearance or translocation to lymph nodes—and absorption to blood. Particle transport is treated as a function of deposition site but independent of particle size and material. For most regions, time-dependent mechanical transport is modeled by considering the region to be made up of several compartments with different clearance half-times. Absorption into the blood depends on the physicochemical form of the radionuclide deposited in the respiratory system, but is taken to be independent of deposition site [32].

This situation is different for gases and vapors, for which deposition in the respiratory tract is material specific. The fraction of an inhaled gas or vapor that is deposited in each region depends on its solubility and reactivity. Gases and vapors

are assigned to three default classes, on the basis of the initial pattern of deposition in the respiratory tract [27]:

- Class SR-0, insoluble and nonreactive: negligible deposition in the respiratory tract (e.g., $^{41}$Ar, $^{85}$Kr and $^{133}$Xe).
- Class SR-1, soluble or reactive: deposition may occur throughout the respiratory tract (e.g., tritium gas, $^{14}$CO, $^{131}$I vapor, and $^{195}$Hg vapor).
- Class SR-2, highly soluble or reactive: total deposition in the extrathoracic airways (e.g., HTO).

The alimentary tract model comprises: (1) oral cavity, including the mouth, teeth, and salivary glands; (2) esophagus; (3) stomach; (4) small intestine including duodenum, jejunum and ileum; (5) large intestine divided into three regions: right colon, left colon, and recto sigmoid. Pancreas and liver are also included in the alimentary tract system [28]. The model represents the following processes:

- Entry of a radionuclide into the oral cavity by ingestion or into the esophagus after mechanical clearance from the respiratory tract;
- Sequential transfer through the oral cavity, esophagus, stomach, small intestine, and segments of the colon, followed by emptying in feces;
- Radionuclide deposition and retention on or between the teeth and return to the oral cavity; deposition and retention in the oral mucosa or walls of the stomach and intestines;
- Transfer from the oral mucosa or walls of the stomach and intestines back into the luminal content or into blood (absorption);
- Transfer from various secretory organs or blood into the content of certain segments of the alimentary tract (secretion).

The extent of absorption of radionuclides in the alimentary tract depends on the chemical properties of the element and the specific chemical form of the intake. Likewise, extent of secretion of systemic activity into the tract is dependent on the chemical form of the element in blood and tissues. The small intestine is the predominant site of absorption, but absorption of some radionuclides can occur in the mouth, stomach, and colon. Fecal excretion, together with urinary excretion, is the main route of radionuclides losses from the body.

Radionuclides can also enter the body through wounds. Although much of the radionuclide may be retained at the contaminated wound site, soluble material can be transferred to the blood and hence to other parts of the body at a rate which depends on their solubility. The NCRP has developed a model to describe this transfer for materials in different physicochemical forms [33]. Coupled with an element-specific systemic biokinetic model, this model can be used to calculate committed doses to organs and tissues and committed effective doses following transfer of the radionuclide to the blood and systemic circulation, as well as to predict urinary and fecal excretion.

Certain materials, e.g., tritiated water in liquid or vapor form, organic carbon compounds and iodine in vapor form or in solution, can penetrate the body trough

intact skin as well. In these cases, a fraction of the activity might enter the blood. There is no general model for absorption of radionuclides through the skin. The basal cells of the epidermis, the skin tissue at radiogenic risk, cannot be represented in the voxel geometry of the reference phantoms. However, a range of 50–100 μm below the skin surface is considered appropriate for specifying the depth of the sensitive layer of most parts of the skin that, in practice, are not protected by clothing. In this case, ICRP recommends to calculate skin doses to sensitive cells that are assumed to be at a depth of 70 μm as a reasonable mean depth of this cell layer [19].

Internal dose calculation standard approaches are not based on data from individual persons, but on a series of reference anatomical and physiological values, and human phantoms [34, 35]. Current anthropomorphic phantoms—like the reference male and female phantoms in Fig. 6.3 and the VIP-Man shown in Fig. 6.6—are mathematical representations based on medical images of real persons. These computational models allow Monte Carlo simulation of radiation transport and energy deposition.

Dosimetric models are the models that describe the energy deposition in the target regions. The quantity to evaluate the equivalent dose to a target tissue (T)—a tissue or organ in which radiation is absorbed—is the committed equivalent dose in a period of 50 years for adults and 70 years for children. The region within the

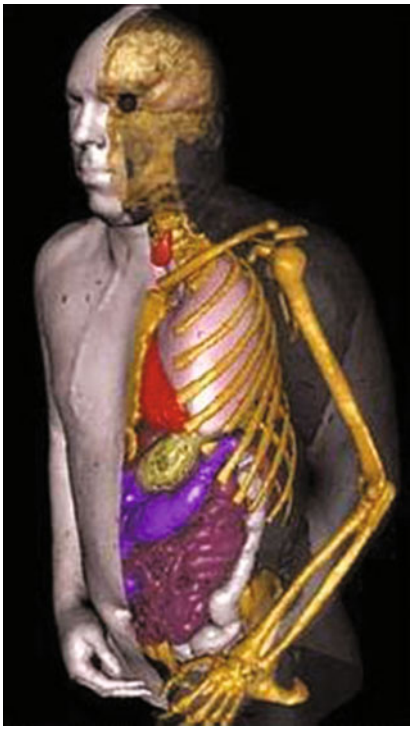

**Fig. 6.6** VIP-Man: phantom developed by Dr. Xu and team at Rensselaer Polytechnic Institute in Troy, NY [This work of art is a Free Art License]

body containing the radionuclide is known as the source region (S). The region may be an organ, a tissue, the contents of the gastrointestinal tract or urinary bladder, or the surfaces of tissues as in the skeleton and the respiratory tract.

The committed equivalent dose to a target tissue T over a period of 50 years can be expressed as

$$H_T(50) = c \cdot \sum_s \sum_j U_{Sj} \cdot SEE(T \leftarrow S)_j, \text{ Sv},$$

where

c = numerical constant required only if a consistent set of units is not employed; e.g., if *SEE (T← S)* is in MeV.$g^{-1}$ per transformation like in ICRP Publication 30, to convert MeV into J, and $g^{-1}$ into $kg^{-1}$, the numerical constant is 1.60. $10^{-10}$;

$U_{Sj}$ = number of nuclear transformations of radionuclide $j$ in source region $S$ during the commitment period following the intake;

$SEE\ (T \leftarrow S)_j$ = specific effective energy deposited in tissue $T$ per nuclear transformation of radionuclide $j$ in source region $S$, expressed as J s/Bq kg or Sv/Bq s

The number of transformations $U_{Sj}$ is given by

$$U_{Sj} = \int_0^t q_{Sj}(t)dt,$$

where $q_{Sj}\ (t)$ is the activity of radionuclide $j$ in source region $S$ at time $t$, assuming a unit activity inhaled, ingested, or injected at time zero. Using a 50-year commitment period $t = 50$.

The specific effective energy representing the equivalent dose in target tissue $T$ per nuclear transformation of a given radionuclide in source region $S$, for the Reference Adult Male and Reference Adult Female, is then

$$SEE(T \leftarrow S)_j = \frac{\sum_j Y_i \cdot E_i \cdot AF_i(T \leftarrow S)_i w_i}{m_T}, \text{Sv}\,(\text{Bq} \cdot \text{s})^{-1},$$

where

$Y_i$ = yield of radiation $i$ per nuclear transformation of radionuclide $j$, $Bq{\cdot}s^{-1}$

$E_i$ = energy of radiation $i$, J

$AF_i(T \leftarrow S)_i$ = fraction of energy absorbed in tissue $T$ per emission of energy $i$ in source region S. It is supposed that, except for mineral bone and GI organs, alphas and betas are completely absorbed in source region S;

$m_T$ = mass of target tissue $T$, of Reference Man, kg

$w_i$ = weighting factor for radiation $i$

Specific effective energies for an adult male and an adult female can be calculated with the revised version of SEECAL, which now utilizes the photon-specific absorbed fractions derived with Monte Carlo methods for the new reference male and female voxel-based phantoms adopted by the ICRP [36].

Committed effective dose is then the sum of committed equivalent doses in target tissues.

$$E(50) = \sum_T w_T \cdot H_T(50) \ldots \mathrm{Sv}$$

For almost all practical assessments, committed effective dose may be estimated using dose coefficients (dose per unit intake, $\mathrm{Sv \cdot Bq^{-1}}$) by inhalation or ingestion. They are the fundamental quantity recommended by ICRP for protection purposes and are presently tabulated in ICRP Publication 119 [21] for a large number of radionuclides, for adults and children of various ages.

## References

1. MCU Project (2015) Monte Carlo modelling of particle transport. http://mcuproject.ru/eabout.html. Accessed 18 April 2015
2. Monte Carlo Code Group (2009) A General Monte Carlo N-Particle (MCNP) transport code, Los Alamos National Laboratory. https://mcnp.lanl.gov/
3. MCNPX (2013) Welcome to the home of the MCNPX code!, Los Alamos National Laboratory. https://mcnpx.lanl.gov/
4. Rensselaer Radiation Measurement & Dosimetry Group (2007–2009) CCHP, The consortium of computacional human phantoms. http://www.virtualphantoms.org/
5. Xu GX, Eckerman KF (2009) Handbook of anatomical models for radiation dosimetry, CRC Press
6. DeWerd LA, Kissick M (2014) The phantoms of medical and health physics: devices for research and development. Springer, New York
7. Hunt J (2015) Visual Monte Carlo Software for radiation protection and internal dosimetry, Radiological Protection and Dosimetry Institute- CNEN Brazil. http://www.vmcsoftware.com/Index.html. Accessed 18 April 2015
8. Hunt JG, Dantas BM et al (2013) Twenty years of visual monte carlo. In: IX latin american irpa regional congress on radiation protection and safety–irpa 2013, Rio de Janeiro
9. NIST, Physical Measurement Laboratory, Physical Reference Data, NIST. http://www.nist.gov/pml/data/
10. Hubbell JH, Seltzer SM (2014) NISTIR 5632. Tables of X-Ray mass attenuation coefficients and mass energy-absorption coefficients from 1 keV–20 MeV for elements Z = 1–92 and 48 additional substances of dosimetric interest, NIST Physical Measurement Laboratory, Updated October 8 2014. http://www.nist.gov/pml/data/xraycoef/index.cfm
11. Unger LM, Trubey DK (1982) Specific gamma-ray dose constants for nuclides important to dosimetry and radiological assessment, TN: Oak Ridge National Laboratory; Report No. ORNL RSIC-45/R1, Oak Ridge
12. Shleien B (1992) The health physics and radiological health handbook, Scinta, Revised Sub edition
13. Ninkovic MM, Adrovic F (2012) Air kerma rate constants for nuclides important to gamma ray dosimetry and practical application, in Gamma Radiation, InTech

14. Wasserman H, Groenewald W (1988) Air kerma rate constants for radionuclides. Eur J Nucl Med 14(11):569–571
15. NPL National Physical Laboratory (2015) Kaye & Laby Tables of Physical & Chemical Constants, National Physical Laboratory. http://www.kayelaby.npl.co.uk/atomic_and_nuclear_physics/4_5/4_5_4.html
16. Eckerman KF, Center for radiation protection knowledge ORNL. http://ordose.ornl.gov
17. RADAR. The radiation dose assessment resource, Stanford Dosimetry, LLC, Beta dose to Skin. http://www.doseinfo-radar.com/RADAR-EXT.html
18. Hamb DM, Mangini CD, Caffrey JA, Tang M (2014) VARSKIN 5: a computer code for skin contamination dosimetry. NUREG/CR-6918 Rev. 2. Nuclear Regulatory Commission, Washington
19. ICRP Publication 116 (2010) Conversion coefficients for radiological protection quantities for external radiation exposures, Annals of the ICRP, vol. 40, no. 2–5
20. ICRP Publication 110 (2009) Adult reference computational phantoms, Annals of the ICRP, vol. 39, no. 2
21. ICRP Publication 119 (2012) ICRP 2012. Compendium of Dose Coefficients based on ICRP Publication 60, Annals of the ICRP, vol. 41, no. Supplement, Supplement 1
22. ICRP Publication 74 (1996) ICRP 1996. Conversion coefficients for use in radiological protection against external radiation, Annals of the ICRP, vol. 26, no. 3/4
23. International Atomic Energy (2014) Radiation protection and safety of radiation sources: international basic safety standards, IAEA safety standards series GSR part 3. IAEA, Vienna
24. ICRP Publication 60 (1991) 1990 Recommendations of the international commission on radiological protection, Annals of the ICRP, vol. 21, no. 1–3
25. Radiation Safety Information Computational Center (RSICC), Oak Ridge National Laboratory. https://rsicc.ornl.gov/Default.aspx
26. International Atomic Energy (2004) Methods for assessing occupational radiation doses due to intakes of radionuclides, Safety Report Series No. 37. IAEA, Vienna
27. ICRP Publication 66 (1994) Human respiratory tract model for radiological protection, Annals of the ICRP, vol. 24, no. 1–3
28. ICRP Publication 100 (2006) Human alimentary tract model for radiological protection, Annals of the ICRP, vol. 36, no. 1–2
29. ICRP Publication 30 (1979–1981, 1988) Limits for intakes of radionuclides by workers, Annals of the ICRP, vols. 2, 4, 6, 19, no. 3–4, 2–3, 4, 1979, 1980, 1981, 1988
30. ICRP Publication 54 (1989) Individual monitoring for intakes of radionuclides by workers, Annals of the ICRP, vol. 19, no. 1–3
31. ICRP Publication 78 (1997) Individual monitoring for internal exposure of workers, Annals of the ICRP, vol. 27, no. 3–4
32. International Atomic Energy, International Labour Office (1999) Safety Guide No.RS-G-1.2. Assessment of occupational exposure due to intakes of radionuclides, IAEA, Vienna
33. NCRP Report No. 156 (2006) Development of a biokinetic model for radionuclide-contaminated wounds and procedures for their assessment, Dosimetry and Treatment, National Council on Radiation Protection and Measurements
34. ICRP Publication 89 (2002) Basic anatomical and physiological data for use in radiological protection: reference values, Annals of the ICRP, vol. 32, no. 3–4
35. Xu XG, Chao TC, Bozkurt A (2000) VIP-Man: an image-based whole-body adult male model constructed from color photographs of the visible human project for multi-particle monte carlo calculations, Health Physics, vol. 78, no. 5
36. Cristy M, Eckerman KF (1993) SEECAL: program to calculate age-dependent specific effective energies, OAK Ridge National Laboratory, ORNL/TM-12351, December 1993

# Chapter 7
# Shielding

Shielding, in conjunction with time and distance, has been all along one of the rules of thumb for protection. Less time means less external exposure. Maximizing the distance from the radiation source lessens the dose.[1] Placing the appropriate shielding between radiation source and individuals further reduces dose from external exposure to acceptable levels.

Biological shielding, that is, a complex of structures and materials to reduce radioactive emissions to a biologically safe level, is one of the most important requirements for reactors, accelerators, irradiators, etc. Structural shielding requirements, including walls, windows, doors, floors, and ceiling, apply to all premises where radiation apparatus (e.g., radiographic or radiotherapy apparatus, etc.) and/or sealed sources are to be used or installed, and radioactive substances are to be used or stored. Additional temporary shielding, including shielding assemblies, lead bricks, paraffin blocks, portable shields, and acrylic and/or lead acrylic radiation shields, etc., is usually required for specific tasks or jobs.

When calculating the thickness of shielding, it is important to consider:

- The type of radiation to shield against, e.g., *X*-rays, gamma, bremsstrahlung, neutrons, accelerated particles, and its energy—the greater the energy the thicker the shield;
- The shielding material to be used, e.g., lead, concrete, steel, paraffin, water, etc.; its shape and geometry.

Absorbing layers—generally designed as primary and secondary barriers—are used to effectively decrease radiation exposure. Primary barriers reduce incident direct radiation. Secondary barriers reduce dose from scattered radiation and equipment leakage.

[1]Dose means equivalent dose, effective dose, ambient dose equivalent, kerma or any other measure used as the case may be.



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Since charged particles typically undergo many collisions along their path and gradually lose their kinetic energy, they are relatively easy to block. Alpha particles can be absorbed by a thin sheet of paper or by 1–2 cm of air. Although beta particles can travel in air a few hundred times farther—up to two meters—common low-energy beta emitters used in laboratories can be shielded by light clothing—laboratory gowns—or a few centimeters of air. Likewise, a centimeter or two of plastic will be enough shielding for higher energy beta emitters.

On the other hand, penetrating radiation—gamma radiation, *X*-rays, and neutrons—are able to travel many meters in air and many centimeters in human tissue, and readily penetrate most materials. They cannot be completely absorbed and need different shielding materials to reduce the dose received. Gamma radiation and *X*-rays require dense shielding materials like lead. Hydrogen-rich materials such as concrete, where a considerable amount of water molecules are chemically bound to the cement, are more effective for combined shielding of both gamma rays and neutrons.

Protective clothing, goggles, and respirators can protect from internal contact with or ingestion of radioactive materials, but provide no protection from gamma or neutron external sources.

## 7.1 Shielding of Photons

When gamma radiation is incident on a finite thickness of material, there is some probability that the radiation interacts in the material and be attenuated.

### *7.1.1 Point Source Shielding*

If considering a narrow or collimated beam of monoenergetic photons imbedded in an infinite medium, with incident intensity $\phi_o$ penetrating a layer of material with mass thickness x, it emerges with intensity $\phi(x)$ given by the exponential expression:$\phi(x) = \phi_0 e^{-\mu x}$, where $\mu$ is the linear attenuation coefficient for the photons of the given energy in the shield material, in $cm^{-1}$, and x is the shield thickness. The linear attenuation coefficient describes the total interaction probability per unit path length, i.e., the fraction of a beam of photons that is absorbed or scattered per unit thickness of the absorber. The larger the value of the attenuation coefficient, the more likely it is that photons of a given energy will interact in a given thickness of material.

The linear attenuation coefficient is obtained multiplying the mass attenuation coefficient for the given energy, $\mu/\rho$ ($cm^2.g^{-1}$) by the shield material mass density, $\rho$ ($g.cm^{-3}$);

$$\mu = (\mu/\rho)\times\rho, \mathrm{cm}^{-1}$$

Updated mass attenuation coefficients and density for a number of materials of interest can be found at the National Institute of Standards and Technology (NIST) website [1].

Once the mass attenuation coefficient is known, the physical thickness, x (cm), of a specific shielding material to reduce the gamma radiation intensity to a desired level can be calculated by solving the shielding equation for x:

$$\mathrm{x} = ln(\phi(\mathrm{x})/\phi_0)/ - \mu\rho$$

With more real broad beam geometry, the use of linear attenuation coefficient alone will generally underestimate the dose rate after shielding; especially for thick shields. To take into account the actual effect of the scattered radiation, a correction factor named the buildup factor is required. The exponential attenuation equation for an isotropic point source will be then,

$$\phi(x) = B(\mu_{ab}x, E)\phi_0 \cdot e^{-\mu x},$$

where $B(\mu_{ab}x, E)$ = buildup factor, a function of the photon energy, the shield material and thickness, source and shield geometry, and the distance from the shield surface to the dose point. For data interpolation and different calculations, the buildup factor is usually expressed in Taylor's form or other analytic function.

$$B(\mu_{ab}x, E) = A_1 e^{-\alpha_1\mu(E)x} + (1 - A_1)e^{-\alpha_2\mu(E)x} \quad \text{(Taylor)},$$

where $\mu_{ab}$ = linear energy absorption coefficient and constants $A_1$, $\alpha_1$, and $\alpha_2$ are tabulated values. Buildup factors can be found in several sources [2–6]. Such values are arranged according to shield material, photon energy, and shield thickness, usually expressed as the product $\mu(E)x$, which represents the number of mean free paths (mfp) or relaxation length. It denotes the average distance traveled in an absorber before an interaction takes place and is the reciprocal of the linear attenuation coefficient:

$$\lambda = 1/\mu$$

Photons are better absorbed by materials with high atomic numbers and high density. The higher the energy of the photons, the thicker the shielding required.

The easiest approach for determining the necessary thickness to block radiation is the use of HVL or HVT (half-value layer or thickness). The HVL is the thickness of a given material required to reduce the radiation dose rate to half of the original or unshielded dose rate. HVL is energy dependent, inversely proportional to the attenuation coefficient, and expressed in units of distance (mm, cm). Table 7.1 shows examples of half-value layers at varying energies for various materials [7].

**Table 7.1** Half-value layers (in cm) for gamma and *X*-ray radiations ["Reprinted from Chap. 3 RSSC 07/11 with permission of University of Florida"]

| Energy MeV | Lead ($\rho$ = 11.35 g/cm$^3$) | Iron ($\rho$ = 7.86 g/cm$^3$) | Aluminum ($\rho$ = 2.82 g/cm$^3$) | Water ($\rho$ = 1 g/cm$^3$) | Air ($\rho$ = 0.0013 g/cm$^3$) | Concrete ($\rho$ = 2.35 g/cm$^3$) |
|---|---|---|---|---|---|---|
| 0.3 | 0.16 | 0.85 | 2.46 | 5.82 | 5.13E + 03 | 2.76 |
| 0.5 | 0.40 | 1.06 | 2.94 | 7.53 | 6.24E + 03 | 3.39 |
| 1 | 0.82 | 1.47 | 4.23 | 9.76 | 8.45E + 03 | 4.65 |
| 1.5 | 1.17 | 1.83 | 5.06 | 12.16 | 1.03E + 04 | 5.72 |
| 2 | 1.36 | 2.07 | 6.19 | 13.86 | 1.24E + 04 | 6.66 |
| 2.5 | 1.44 | 2.29 | 6.79 | 15.75 | 1.39E + 04 | |
| 3 | 1.47 | 2.48 | 7.37 | 17.77 | 1.51E + 04 | 8.15 |
| 3.5 | 1.51[a] | 2.59 | 7.97 | 19.25 | 1.65E + 04 | |
| 4 | | 2.76 | 8.35 | 20.38 | 1.78E + 04 | 9.36 |
| 5 | | 2.85 | 9.63 | 23.10 | 2.04E + 04 | 10.34 |
| 10 | | 2.95[a] | 11.75 | 30.13 | 2.48E + 04 | 13.86 |
| 20 | | | 12.16[a] | 38.50 | 3.30E + 04 | 14.14[a] |
| 30 | | | | 40.76 | 3.36E + 04 | |
| 40 | | | | 43.31[a] | 34650[a] | |

[a]Value beyond which the half-value layer (HVL) will not increase regardless of increase in energy

HVL can be calculated by setting $I_x/I_0 = \frac{1}{2}$ and solving the attenuation equation for x:

$$1/2 = e^{-\mu x_{1/2}}$$
$$x_{1/2} = \frac{ln(2)}{\mu}$$
$$x_{1/2} = \frac{0.693}{\mu} = HVL$$

In a similar manner, a tenth-value layer or thickness (TVL or TVT) is defined as the thickness that will reduce the radiation intensity to one-tenth of the unshielded value. The reduction of n TVL layers is $10^n$.

$$x_{1/10} = \frac{ln(10)}{\mu} = TVL$$

Since dose rate is the result of the absorption and scattering of individual photon energies, when performing shielding calculation all yields and energies of the given radionuclides have to be accounted for. If energies are close to each other, they can be grouped using the average energy and combined yields.

### 7.1.2 Extended Source Shielding

The concept of extended source covers all sources—line, surface, and volume sources—whose dimensions cannot be disregarded in shielding calculations. The source is then represented by an infinite number of elementary cells—point kernels—distributed throughout the source dimensions, and the contributions from all elements are added up by numerical integration to obtain the dose rate for the total source volume.

$$\dot{D}_{(x)} = \int_V \frac{A_V(V) \cdot \Gamma}{r^2} \cdot B(z, \mu x) \cdot e^{-\mu x} dV,$$

where

$\frac{A_V(V) \cdot \Gamma}{r^2}$ = dose rate contribution from each differential element of source;
$B(z, \mu x)$ = kerma buildup factor for the given shield material z;
$\mu$ = lineal attenuation coefficient for the shield material; and
$x$ = photon path length through a shield elementary volume dV

Assumptions for the point-kernel method are that both the source and shielding are isotropic and the medium has the same attenuating properties in all directions. Attenuation by increasing distance from the source as well as exponential attenuation and scattering of the photons are taken into consideration.

Point-kernel methods have the advantage to be very fast and relatively easy. Some available commercial software for shielding calculation is based on the point-kernel method.

### 7.1.3 Shielding Against Scattered Radiation

If the source is incident on a thick barrier, backscattered photons will produce a dose angular distribution or material albedo directed away from the shielding slab. In this interaction, the photon rebounds with an energy which is directly dependent on the scattering angle and the incident energy, thus increasing the albedo as the source energy decreases. The dose rate from scattered radiation is:

$$\dot{H}_s = \frac{\dot{H}_R \cos\alpha \cdot a(E_0, \alpha) \cdot A_R}{2\pi L^2},$$

where

$\dot{H}_S$ = dose rate due to scattering at a distance $L$ from the scattering surface area, J.kg$^{-1}$·h$^{-1}$;
$\dot{H}_R$ = dose rate from the useful beam at the scattering surface, J.kg$^{-1}$·h$^{-1}$;
$\alpha$ = angle of radiation incident with respect to the surface normal;
$a(E_0,\alpha)$ = differential dose albedo (coefficient of dose reflection) for incident energy $E_0$ at angle $\alpha$;
$A_R$ = area of useful beam at scattering surface, m$^2$; and
$L$ = distance from the scattering surface area, m

Table 7.2 shows some examples of differential dose albedo, $a\ (E_0,\alpha)$, for ordinary concrete [8].

Scattered radiation energy $E'$ can be obtained from the following expression:

$$E' = \frac{E_0}{1 + E_0/mc^2},$$

where $mc^2$ is electron rest energy = 0.511 MeV.

Reflected radiation may be of importance to determine, for example, the shielding design of maze-protected doors and around ducts; also, to evaluate the scatter of the primary beam to the secondary barrier for high-energy machines and radiation that goes over the shield and scatters in the air toward the ground.

**Table 7.2** Differential dose albedo for ordinary concrete [8]

| cos α | E (MeV) | | |
|---|---|---|---|
| | 0.1 | 0.661 | 1.25 |
| 0.0 | 0.52 | 0.38 | 0.32 |
| 0.1 | 0.45 | 0.29 | 0.24 |
| 0.2 | 0.37 | 0.23 | 0.18 |
| 0.3 | 0.33 | 0.19 | 0.14 |
| 0.4 | 0.29 | 0.16 | 0.11 |
| 0.5 | 0.27 | 0.13 | 0.08 |
| 0.6 | 0.23 | 0.11 | 0.08 |
| 0.7 | 0.22 | 0.09 | 0.06 |
| 0.8 | 0.20 | 0.08 | 0.05 |
| 0.9 | 0.18 | 0.07 | 0.04 |
| 1.0 | 0.17 | 0.06 | 0.04 |

### *7.1.4 Commercial Software*

Shielding calculations done manually are often rough approximations. The most accurate shielding calculations are performed by computers. Here are some examples of commercial software that can be used for photon dose and shielding calculation. The RadPro Calculator; a free online tool that performs many calculations, including gamma dose rate from a point source, with or without shielding; can also be used [9].

- MicroShield—a comprehensive photon/gamma-ray shielding and dose assessment program from Grove Software Inc., widely used for designing shields and estimating dose from different gamma sources geometries—point, line, disk, sphere, infinite plane, annular cylinder, etc. Latest version uses conversion coefficients from ICRP Publication 116 [10] consistent with ICRP 2007 recommendations [11].
- RANKERN—a program for gamma-ray transport solutions, which provides a rapid method for the design and assessment of gamma-ray shielding and dosimetry. RANKERN uses the point-kernel technique coupled with buildup factors, and is suitable for determining dose rates through shield materials, in ducts and labyrinths and from skyshine.
- MERCURAD—simulation software from Canberra using Monte Carlo method for numerical integration that can display 3D scenes for dose rate and shielding calculations. The latest version integrates a powerful function for buildup factor calculation.

## 7.2 *X*-Ray Shielding

The simplest way to calculate *X*-ray shielding is using the technical characteristics given by the equipment manufacturer. Knowing the tube type and voltage of the *X*-ray generator, and the material and thickness of inherent filtration, it is easy to find the required shielding from existing tables and nomograms [3, 12, 13].

The type of equipment also determines where the *X*-ray beam will be directed, the number and type of procedures to be performed, the location of the radiographer, and the energy (kVp) of the *X*-rays.

To obtain the barrier transmission factor, $B$, i.e., the amount of radiation passing through barrier at a certain distance, $d$, from the source as a result of primary radiation, scatter radiation, and leakage radiation, the following equation can be used [14];

$$B = Pd^2/WUT,$$

where

$P$ = maximum permissible dose rate (kerma value) for controlled and noncontrolled areas, usually 1 mSv/week for radiographers, radiologists, and other radiation workers, and 0.02 mSv/week for members of the public;

$W$ = workload, Gy/week;

$U$ = use factor;

$T$ = occupancy factor, the average fraction of time that a particular place is occupied by staff, patients, or public when the *X*-ray beam is on

Presently, NCRP-147 [12] utilizes three shielding models for *X*-rays:

- Model No. 1—an extension of the above method based on NCRP 49, with new models for image receptor attenuation and leakage, and a computer program to implement fully. XRAYBARR [15] is a program to calculate the thickness of lead, concrete, gypsum, steel, plate glass, and wood required to shield diagnostic *X*-ray installations using W and Mo anode *X*-ray tubes in the range of 25–150 kVp.
- Model No. 2—based on data from model No. 1, the NCRP 147 shows kerma per patient at 1 m and transmission curves for a given workload; with it the unshielded kerma can be calculated and then the transmission factor needed to reduce to P/T;

$$K_0 = K^1 UN/d^2,$$

where

$K^1$ = average kerma—primary and secondary—expected for a patient procedure;

$U$ = use factor, replaced by 1 for secondary barriers;

$N$ = patient procedures per week;

$d$ = distance

**Table 7.3** NCRP 147 Recommended Occupancy Factors ["Reprinted with permission of the National Council on Radiation Protection and Measurements, http://NCRPpublications.org"]

| T | Type of area |
|---|---|
| 1 | Offices, labs, pharmacies, receptionist areas, attended waiting rooms, children play areas, *X*-ray rooms, film reading areas, nursing stations, *X*-ray control rooms |
| 1/2 | Patient examination and treatment rooms |
| 1/5 | Corridors, patient rooms, employee lounges, employee rest rooms |
| 1/8 | Corridor doors |
| 1/20 | Public toilets, vending machine areas, storage rooms, outdoor areas w/seating, unattended waiting rooms, patient holding areas |
| 1/40 | Outdoor spaces, unattended parking lots, attics, stairways, unattended elevators, maintenance personnel closets |

The acceptable thickness, x, of a shielding barrier will be that which provides transmission, B, not in excess of:

$$B(x) = P/T/K_0$$

A conservative, realistic model will include any preshielding provided by the image receptor (cassette, cassette holder) and the table, thus: $x_{tot} = x + x_{pre}$ [16].

- Model No. 3—NCRP-147 shows barrier thickness requirements as function of $NT/Pd^2$, in $mGy^{-1} \cdot m^{-2}$, calculated for representative rooms. These include Radiographic Rooms, Radiographic and Fluoroscopic Rooms, and Cardiac Cath Labs.

NCRP 147 recommended occupancy factors are shown in Table 7.3 [17].

NCRP 147 is focused on standardizing the structural shielding for *X*-ray imaging devices [12]. It contains 12 $NT/Pd^2$ graphs for representative Radiographic and R&F Rooms—primary barriers with preshielding, primary barriers without preshielding and secondary barriers for lead and concrete—also, thickness requirements for steel, gypsum wallboard, and glass; while AAPM Task Group 108 [18] and NCRP 151 [19], respectively, address shielding for *X*-ray from PET/CT facilities and from megavoltage *X*-ray radiotherapy.

NTPd2 is a Windows XP-based program to also calculate radiation shielding requirements for diagnostic radiology facilities following NCRP 147 [15]. The program calculates $NT/Pd^2$ values based on the user's input.

## 7.3 Shielding from Beta Radiation

Beta particles and charged particles will lose their energy by penetrating the shielding absorber due to their small mass and relative high energy; however, since the absorption in the shielding material produces bremsstrahlung radiation,

**Table 7.4** Thicknesses to absorb beta radiation

| Energy (MeV) | Plastic (Lucite) | Concrete | Aluminum |
|---|---|---|---|
| 0.5 | 0.3 | 0.1 | 0.1 |
| 1.0 | 0.5 | 0.3 | 0.3 |
| 2.0 | 0.8 | 0.5 | 0.5 |
| 3.0 | 1 | 0.8 | 0.8 |

bremsstrahlung has to be considered when shielding from beta emitters and charged particles. The fraction of beta energy reappearing as bremsstrahlung is approximately ZE/3000 where Z is the atomic number of the absorber and E is the beta energy in MeV. In case of beta plus decay, photons from electron–positron annihilation have also to be considered.

The thickness and choice of shielding material depends upon the stopping power of the material for the highest energy beta and any photon (bremsstrahlung) production. Safety glasses used in laboratories generally absorb 90 % of high-energy beta particles ($\sim$1 MeV). Shielding from beta particles and charged particles of specific energy is determined by its range ($R$) for that energy in g/cm$^2$ (thickness in cm multiplied by the volume density of the material in g/cm$^3$), which is approximately equal to the maximum energy ($E_{max}$) in MeV divided by 2 ($R = E_{max}/2$) [20]. Table 7.4, courtesy of the Handford ALARA Center [21], provides some examples of thicknesses in cm for materials commonly used to absorb beta radiation.

When performing procedures that involve annihilation radiation (e.g., PET procedures) considerations must be taken to shield against broad beams of high-energy radiation; materials with high atomic numbers and high density are preferred [18].

## 7.4 Neutron Shielding

Neutrons, like gamma rays, are a highly penetrating form of radiation. Neutron interaction with matter is a highly complex process, which involves both an energy spectrum of neutrons and different types of nuclear reactions, followed by the production of several secondary particles. Neutron interactions with shielding for common energies are achieved through elastic and inelastic scattering. Scattering processes reduce neutron energies and excite material nuclei until neutrons are absorbed by radiative capture, resulting in the immediate release of a substantial amount of high-energy gamma photons [22]. Therefore, shielding neutrons includes calculations for primary neutron flux and for secondary gamma radiation.

To reduce the intensity of a narrow beam of neutrons, it is possible to use an exponential equation similar to that used for monoenergetic photons, but with the attenuation coefficient replaced by the total macroscopic cross section $\Sigma_t$ for the neutron interactions in the given medium;

$$\phi(x) = \phi_0 e^{-\Sigma_t x}$$

The energy-dependent neutron cross section is the interaction probability per unit atom density and distance in $cm^2$.

The concepts of the attenuation coefficient and the half-value layer used for gamma photons also apply to the attenuation of neutrons. Neutron shielding calculations for small neutron sources can be performed using the transmission factor in a manner very similar to that used for gamma shielding. The transmission factor is defined as the ratio of the neutron ambient dose equivalent values with and without shield.

$$k = e^{d/\lambda},$$

where

$\lambda$ = relaxation length for the shielding material and the average energy of neutrons, cm;
$d$ = thickness of shielding, cm

The relaxation length is the thickness of shield that will attenuate a narrow beam of neutrons to 1/e (about 37 %) of its original intensity. As discussed before, relaxation length is numerically equal to the reciprocal of the linear attenuation coefficient $1/\mu$ for gamma radiation or $1/\Sigma_t$ for neutron radiation. Due to the energy spectrum of neutrons, the relaxation length would not be constant while penetrating shielding material. Thus, it is important to know the conditions under which the shielding was calculated, e.g., neutron deep penetration, ranges of neutron source intensities, etc.

Light elements are best for slowing down neutrons and thus, materials with high hydrogen content like water, paraffin, polyethylene, and concrete, are commonly used for neutron shielding. To avoid most of the production of penetrating gamma radiation from low and intermediate neutrons, a combination of polyethylene with boron is used. By a nuclear reaction with neutrons, boron produces alpha particles and residual nuclei which are easily stopped in very small thickness of material.

Neutron shielding calculations are best done by computers. However, simple conservative formulas for attenuation and buildup factors can be used for routine tasks where rough calculation is accepted. For example, secondary gamma radiation from neutron capture in shielding material can be assumed as from a homogeneous volumetric source or, merely, as all photons were originated at a point located in the center of shielding. Thus, the number of photons emitted per second with energy $E_i$ can be consequently estimated by the following;

$$N_{\gamma E_i} = A_E \frac{\sigma_\gamma E_i}{\Sigma_{aT}},$$

where

$A_E$ = neutron source emission rate, $s^{-1}$;
$\sigma_{\gamma E_i}$ = macroscopic absorption cross section for the type of nuclear reaction where a photon of energy $E_i$ is originated, $cm^2$;
$\Sigma_{aT}$ = total macroscopic absorption cross section, $cm^2$

The required shielding is then evaluated finding all contributions for multiple energies.

Nuclear reactor neutron and secondary gamma shielding calculations are complex tasks needed for numerical methods and computational systems, including MCNP (a Monte Carlo N-Particle code) [23], FLUKA (a fully integrated particle physics Monte Carlo simulation package) [24], and SCALE (Standardized Computer Analyses for Licensing Evaluation) [25]. Attenuation of neutrons and production of induced photons can be calculated using the coupled neutron/photon transport code TART2012 developed at Lawrence Livermore National Laboratory (LLNL) [26].

Finally, when shielding particle accelerators, it is important to consider all types of secondary radiation emitted (neutrons, protons, photons, and electrons).

## References

1. NIST (2015) NIST physical measurement laboratory, physical reference data, X-Ray mass attenuation coefficients.The National Institute of Standards and Technology (NIST), 8 Oct 2014. http://www.nist.gov/pml/data/xraycoef/index.cfm. Accessed 26 March 2015
2. Shultis KJ, Faw RE (2000) Radiation shielding. American Nuclear Society
3. American National Standard (1991) Gamma-ray attenuation coefficients & buildup factors for engineering materials. ANSI/ANS-6.4.3
4. Trubey DK (1966) A survey of empirical functions used to fit gamma-ray buildup factors. ORNL, Tennessee
5. Shimizu A, Onda T, Sakamoto Y (2004) Calculation of gamma-ray buildup factors up to depths of 100 mfp by the method of invariant embedding, (III) generation of an improved data set. J Nucl Sci Technol 41(4):413–424
6. Mashkovich VP, Kudriatseva AV (1995) Zaschita at Ioniziruyushij Izluchenya Spravochnik (Handbook Shielding from Ionizing Radiation), 4th edn. Energoatomizdat, Moscow
7. University of Florida Environmental Health and Safety (2015) Radiation safety short course (RSSC). http://www.ehs.ufl.edu/programs/rad/rssc/
8. Cornejo, N (1998) Capítulo VI. Métodos de cálculo de dosis. In: Protección Radiológica en la Aplicación de las Técnicas Nucleares. Havana, Centro de Información de la Energía
9. McGinnis R, Rad pro calculator. http://www.radprocalculator.com/Index.aspx
10. ICRP Publication 116 (2010) Conversion coefficients for radiological protection quantities. Ann ICRP, 40(2−5)
11. Publication ICRP (2007) 103, The 2007 recommendations of the international commission on radiological protection. Annals of the ICRP 37(2):4
12. NCRP Report No. 147, (2004) Structural shielding design for medical X-Ray imaging facilities, Bethesda, 2004, Revised 2005

13. Jaeger RG et al (1968) Engineering compendium on radiation shielding, vol 1: shielding fundamentals and methods, New York-Berlin: Springer
14. NCRP Report No. 49 (1976) Structural shielding design and evaluation for. medical use of X-rays and gamma-rays up to 10 MeV, National Council on Radiation Protection and Measurements, Bethesda
15. Simpkin DJ, Diagnostic X-ray shielding, DSimpkin Medical Physics. https://sites.google.com/site/dsimpkinmedicalphysics/home/shielding
16. Dixon RL, Simpkin DJ (1998) Primary shielding barriers for diagnostic X-ray facilities. Health Phys 74(2):181–189
17. Simpkin DJ (2007) CRSO Meeting, Portland, OR 7/10/07 NCRP 147 Shielding Calculations, Portland
18. Madsen MT, Anderson JA, Halama JR et al (2006) AAPM task group 108: PET and PET/CT shielding requirements, Med. Phys, vol. 33, no. 1
19. NCRP Report No. 151 (2005) Structural shielding design and evaluation for megavoltage X- and gamma-ray radiotherapy facilities, Bethesda
20. Mattsson S, Hoeschen C (2013) Radiation protection in nuclear medicine, Springer
21. Flour Handford ALARA Center (2007) Report for August 27 2007 Attachment 1
22. Shapiro J (2002) Radiation protection: a guide for scientists, regulators and physicians, Fourth Edition, Harvard University Press
23. Los Alamos National Laboratory, A general monte carlo n-particle (MCNP) transport code, U. S. Department of Energy. https://mcnp.lanl.gov/
24. CERN, INFN, "FLUKA," European organization for nuclear research, national institute for nuclear physics. http://www.fluka.org/fluka.php
25. ORNL, SCALE a comprehensive modeling and simulation suite for nuclear safety analysis and design, Radiation Safety Information Computational Center (RSICC), Oak Ridge National Laboratory. http://scale.ornl.gov/index.shtml
26. Cullen DE (2012) Report: LLNL-TR-577352 TART2012 an overview of a coupled neutron-photon 3-D, combinatorial geometry time dependent monte carlo transport code. Lawrence Livermore National Laboratory, Livermore

# Chapter 8
# Exposure Situations

Radiation sources, meaning by them "anything that may cause radiation exposure—such as by emitting ionizing radiation or by releasing radioactive substances or material" [1] ["Reproduced with permission by the IAEA."]—can be man-made sources—a radiation generator, or sealed radioactive source, or radioactive material—facilities and activities, and any other practice or circumstances in which people may be exposed to radiation from naturally occurring or artificial sources—and the natural sources themselves—the sun and stars (cosmic radiation) and rocks and soil (terrestrial sources of radiation).

Facilities includes: nuclear facilities; irradiation installations; some mining and raw material processing facilities such as uranium mines; radioactive waste management facilities; and any other places where radioactive material is produced, processed, used, handled, stored, or disposed of—or where radiation generators are installed—on such a scale that consideration of protection and safety is required. Activities includes: the production, use, import, and export of radiation sources for industrial, research, and medical purposes; the transport of radioactive material; the decommissioning of facilities; radioactive waste management activities such as the discharge of effluents; and some aspects of the remediation of sites affected by residues from past activities [1] ["Reproduced with permission by the IAEA"].

Nowadays, radiation sources are common and permanent elements of our lives; thus, the risks associated with radiation exposure can only be restricted, not entirely eliminated. Everybody knows that radiation sources are an essential part of modern-day health care systems, research and development institutions, and manufacturing processes. The benefits that have been obtained from nuclear techniques and nuclear energy since the discovery of radioactivity are undeniable.

While taking full advantage of its benefits—for both society and economic—the responsible use of radiation sources means that it is based upon strictly regulations and the systematically application of a set of measures to ensure the main objective of radiation safety, i.e., avoid the occurrence of harmful tissue reactions (deterministic effects) and keep the likelihood of incurring in exposures (where these are



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not certain to be received), the number of people exposed, and the magnitude of individual doses as low as reasonably achievable [2].

To better understand all actions that encompass the term "responsible use of radiation," it is necessary to first address some concepts that are part of the radiological protection system's current scope.

Instead of the protection approach relying on the distinction between practices and interventions from 1990 ICRP recommendations [3], most recent ICRP Recommendations [2] suggest a different approach based on the distinction between three types of situation—planned, emergency and existing—in which radiation exposure may occur and either the source of exposure or the pathways leading to the doses received by individuals can be controlled by some reasonable means. Note the importance of directly controlling exposure in this approach.

The term practice, as a closely related term to planned exposure situation, is still used to denote an activity that causes an increase in exposure to radiation or in the risk of exposure to radiation [4]. The term intervention is now limited to the protective actions that reduce exposure in case of emergency.

## 8.1 Types of Exposure Situation

The distinction between types of exposure situation is for the purpose of establishing practical requirements for protection and safety as it is stated in the International Basic Standards [4].

- A planned exposure situation is a situation of exposure that arises from the planned operation of a source or from a planned activity that results in an exposure due to a source. In planned exposure situations, provisions for protection and safety can be made in advance and the associated exposures and their likelihood of occurrence can be restricted from the beginning. They involve the deliberate introduction and operation of sources, including design and decommissioning of facilities and equipment, operating procedures and transport, as well as disposal of radioactive waste. Planned exposure situations also cover medical exposure of patients—including their comforters and caregivers—and occupational exposures in connection with existing and emergency exposure situations. Individual dose limitation, and dose and risk constraints apply to planned exposures, except those involving medical exposure of patients.
- An emergency exposure situation is a situation of exposure that occurs as a result of an accident, a malicious act, or any other unexpected event, and requires prompt action to avoid or to reduce adverse consequences. Preventive measures and mitigation actions have to be considered before an emergency exposure situation arises, but exposures can be reduced only by implementing protective actions. Reference levels apply to emergency exposure situations. Once an emergency situation has occurred, they act as benchmark for assessing the effectiveness of protective actions.

- An existing exposure situation is a situation of exposure that already exists when a decision on control has to be taken. They include exposure to natural background radiation, to residues from past practices that were not subject to the current regulatory control, as well as prolonged exposure situations after emergencies. Examples of existing exposure situations are radon in dwellings or the workplace, and contaminated territories arising from an accident after an emergency has been declared to be ended (Chernobyl, Fukushima). Reference levels apply to existing exposure situations.

## 8.2 Dose Constraints and Reference Levels

Dose constraints and reference levels are used for optimization of protection and safety with the objective to control all exposures to levels that are as low as reasonably achievable (ALARA).

Dose constraints are usually set up as a fraction of the dose limit; can be in units of individual dose, collective doses, ambient dose rate, etc., and are selected by regulators and radiation protection administrators at any level based on good practice and on what can reasonably be achieved. It is important to say that dose constraints are not dose limits, neither new "standards of care" for workers; they are just regulatory benchmark values for the retrospective assessment of planning and optimization of protection or select values appropriate only for particular circumstances, including the nature of the exposure and the practicability of its reduction or prevention.

Reference levels are used for optimization of protection and safety in emergency exposure situations and in existing exposure situations. A reference level serves as a boundary condition in identifying the range of options for the purposes of optimization in implementing protective actions. For an emergency exposure situation or an existing exposure situation, it represents the level of dose, risk or activity concentration above which it is judged to be inappropriate to plan to allow exposures to occur, and below which the optimization of protection and safety is implemented [4]. The values chosen for reference levels also depend on the prevailing circumstances for the exposures under consideration, including the net benefit of avoiding preventive or protective actions that would be detrimental to living conditions.

Given that dose constraints and reference levels are not dose limits, exceeding them does not represent any regulatory violation.

In *X*-ray medical imaging, image-guided interventional procedures and diagnostic nuclear medicine, a diagnostic reference level is used to indicate whether, in routine conditions, the dose to the patient or the amount of radiopharmaceuticals administered in a specified radiological procedure for medical imaging is unusually high or unusually low for that procedure.

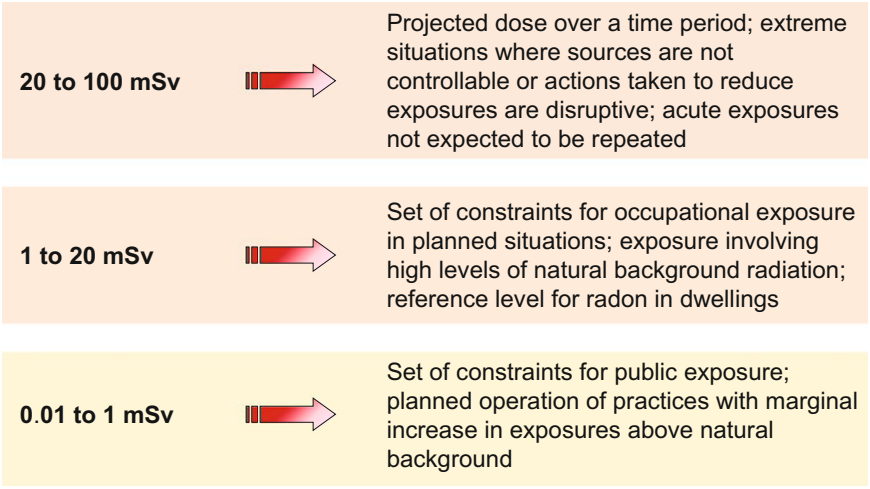


**Fig. 8.1** Set of constraints and reference levels for optimization recommended by ICRP

Currently, the ICRP recommends [2] the selection of constraints in the range up to 1 mSv for situations in which individuals may be exposed to a source that gives them little or no individual benefit, but for which there may be benefits to society in general, such as constraints for public exposure in planned exposures. In such cases the dose constraint would represent a marginal increase, up to about 1 mSv, above the dose received in a year from the natural background radiation.

Where individuals are exposed to sources that are not under control or where actions to reduce doses would be disproportionately disruptive, in unusual and often extreme situations, the ICRP recommends a range of reference levels from 20 to 100 mSv. The ICRP also recommends that the range of doses between these two extremes (1 to 20 mSv) can be used for situations in which individuals usually receive benefit from the exposure situation, but not necessarily from the exposure or the source of exposure itself. Examples are constraints set for occupational exposures and abnormally high levels of background radiation.

The set of specified arrays for constraints and reference levels for optimization purposes recommended by ICRP [2] in different exposure situation are summarized in Fig. 8.1.

## 8.3 Types of Assessment

Provided that exposure to radiation is a consequence of the combination of various events and situations related to one or more sources, e.g., natural background, consumer products, medical applications, occupational exposure, etc., in the assessment of safety and protection, there can be two valid approaches:

- A source-related approach when various individuals are exposed to a source or group of sources and the purpose of the assessment is to assure the optimum protection, and
- An individual-related approach when is the same individual who may be exposed to several sources, and the goal of the assessment is the total exposure.

Figure 8.2 exemplifies the source-related approach, which allows to judge whether or not a particular source or group of sources would give enough benefit; in other words, societal and economic benefits are higher than the detriment they could cause; and furthermore, if all taken actions guarantee the best level of protection achievable under the prevailing circumstances, along with the minimal exposure to all individuals.

The source-related approach sorts out the sum of events and situations which could lead to the exposure of the representative person of present and future populations associated to a specific source or group of sources. Source-related principles apply to all exposure situations regardless the type of source or exposed individual. For planned exposure situations, the source-related restriction to the dose that individuals may incur is the dose constraint. For potential exposures, the corresponding concept is the risk constraint. For emergency and existing exposure

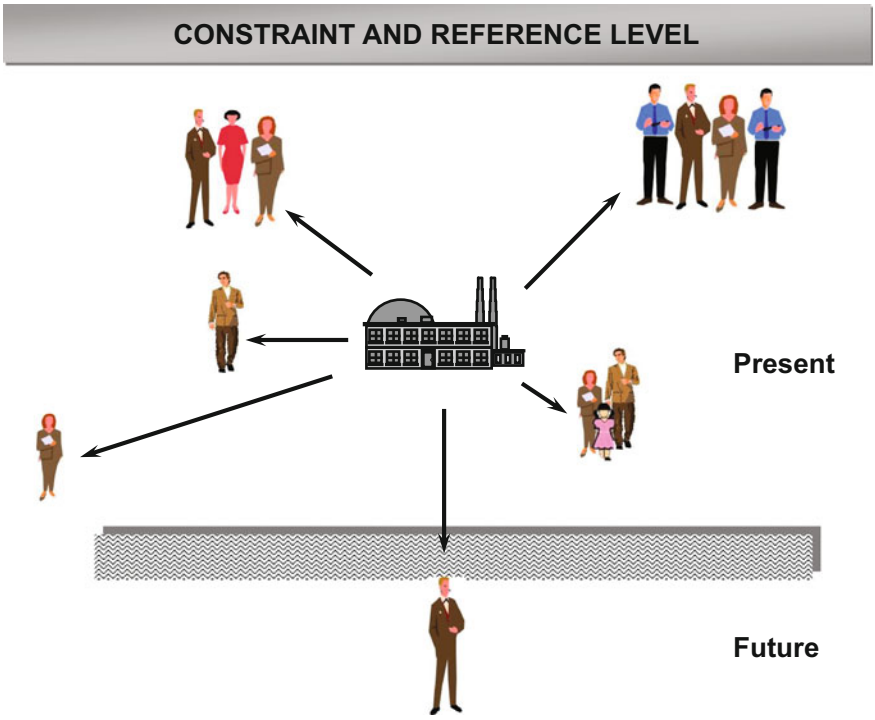


**Fig. 8.2** Source-related approach for the assessment of a single source in all exposure situations

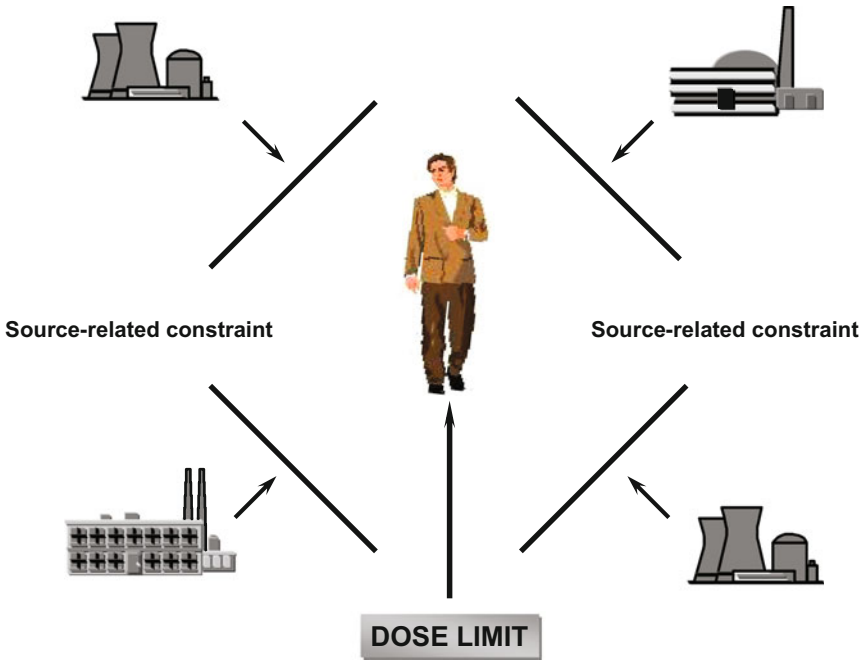


**Fig. 8.3** Individual-related approach for the assessment of all regulated sources in planned exposure situation

situations, the source-related restriction is the reference level. Dose constraints and reference levels are used in conjunction with optimization of protection to assure that all exposures are kept as low as reasonable achievable (ALARA).

Individual-related approach is illustrated in Fig. 8.3. Note that the dose constraint, established for each particular source according to the previous approach, should ensure that the sum of doses from planned operations for all sources under control remains within the dose limit.

The recommendation is that all relevant sources should be considered, but excluding local background radiation. However, it is rarely possible to assess the total exposure of an individual from all such sources and, especially, in the case of public exposure. It will normally be sufficient to focus on the dose received from the source or group of sources likely to dominate.

## 8.4 Exposure Categories

Safety and protection requirements in all exposure situations are also applied to three categories of exposure to individuals: occupational exposure, public exposure and medical exposure [4].

- Occupational exposure is the exposure of workers incurred in the course of their work as a result of reasonable expected circumstances, regardless if it is at a hospital, nuclear plant, factory, research center, or any other kind of facility, with the exception of excluded exposures and exposures from exempt practices or exempt sources. The exposure of workers in certain special conditions, e.g., to natural sources, specifically if they lead to exposure to radon, to technologically enhanced naturally occurring radioactive materials (NORM), and due to high altitude or spatial flights, are also considered occupational exposures. The exposed individuals in this category are workers, i.e., any person who is employed full-time, part-time, or temporarily by an employer, and who also has rights and duties related to radiological protection.
- Medical exposure is the exposure incurred by patients, as part of their own medical or dental diagnosis or treatment; by persons, other than those occupationally exposed, knowingly exposed while voluntarily helping in the support and comfort of patients; and by volunteers in a program of biomedical research involving their exposure. Exposed individuals in this category are patients—in this case, persons who receive an exposure associated to a diagnostic, screening, interventional, or therapeutic procedure which will benefit him or herself—comforters and caregivers, and volunteers. Since the main objective of protection in this category is to deliver the minimum necessary dose to fulfill the clinical purpose of the procedure, the use of diagnostic reference levels is recommended. A dose constraint is used to optimize the protection and safety of caregivers and comforters.
- Public exposure is the exposure incurred by members of the public due to sources in planned exposure situations, emergency exposure situations and existing exposure situations, excluding any occupational exposure or medical exposure. Public exposure may result from liquid and airborne discharges or accidental releases from facilities, residuals from past activities, and the increase of the exposure to natural background attributable to consumers' products and medical uses of radiation. The individuals in this category are members of the public, i.e., any representative person[1] who receives an exposure which is neither occupational, nor medical. Considering that the level of protection to be afforded must be the same to that of a member of the public, the embryos/fetus is also considered in this category.

In planned exposure situations, occupational and public exposures at some level—usually a fraction of the limit—can be expected to occur. An exposure that is reasonably expected to occur is known as normal exposure. If exposure is not expected to occur, but could result from an accident or from an event or a sequence of events that may occur, but is not certain to occur, the exposure is referred to as a

[1]A person whose habits (food consumption, breathing rate, location, usage of local resources) are typical of a small number of individuals representative of those most highly exposed in a given scenario.

**Table 8.1** Summary of exposure classification

| Criterion | | | | |
|---|---|---|---|---|
| Position of the source | Likelihood of occurrence | Individual exposed | | Situation in which exposure may occur |
| External | Normal | Occupational medical public | ➡ Workers<br>➡ Patients<br>➡ Representative person | Planned exposures<br>Emergency exposures<br>Existing exposures |
| Internal | Potential | | | |

potential exposure. Potential exposures may arise following deviations from planned operating procedures, incidents or accidents, and malevolent events.

Deviations from planned exposure situations and likely incidents can be often foreseen and their probability of occurrence estimated at the planning stage of the exposure situation, but cannot be predicted in detail. Loss of control of radiation sources and malevolent events are less predictable. The acceptability of potential exposures is then based on both the probability of occurrence of the exposure and its magnitude. The types of events usually covered are [2]: ["Reprinted from ICRP Publication 103 with permission of the ICRP"].

- Events that would primarily affect individuals subject to planned exposures; the number of individuals is usually small, and the detriment involved is the health risk to the directly exposed persons.
- Events where potential exposures can affect a larger number of individuals; these events not only involve health risks but also other detriments, such as contaminated land and the need to control food consumption.
- Events in which the potential exposures could occur far in the future and the doses are delivered over long-time periods. These events carry on considerable uncertainties.

Table 8.1 summarizes the different types of exposure according to different criteria.

## References

1. International Atomic Energy (2007) IAEA safety glossary, terminology used in nuclear safety and radiation protection—2007 Edition. IAEA, Vienna
2. ICRP (2007) The 2007 recommendations of the international commission on radiological protection. Ann ICRP 37(2−4), April-June 2007
3. ICRP Publication 60 (1991) 1990 recommendations of the international commission on radiological protection. Ann ICRP 21(1−3)
4. International Atomic Energy (2014) Radiation protection and safety of radiation sources: international basic safety standards, IAEA safety standards series GSR Part 3. IAEA, Vienna

# Chapter 9
# Regulations and Regulatory Control

To control radiological protection and safety of radiation sources; to provide for the instruments to enforce compliance with national and international requirements; and to maintain the appropriate standards for protection is required a national infrastructure. This infrastructure consists of at least a legal framework (instruments), a regulatory authority, and a management system, which may include a number of national bodies and persons—institutions—connected with protection and safety [1]. According to the IAEA Safety Standards, the framework for safety must be established for the entire range of facilities and activities, from the use of a limited number of radiation sources to a nuclear power program [2].

## 9.1 Legal and Regulatory Framework

The legal framework supporting radiation protection and safety includes a whole set of legal and regulatory instruments—acts, codes, laws, ordinances, rules, regulatory guides, standards, guidelines, communications, etc.—which governments use to achieve the desired effect.

Both the government and the regulatory authorities have important responsibilities in establishing the legal framework, including establishing standards, subject to a graded approach consistent with the radiation risks associated with the existing facilities and activities [3]. Figure 9.1 gives a general idea of the hierarchy and contents of the legal framework that is part of a regulatory control program, and some examples of the most important regulations in the United States.

The first and more general group covers national or federal instruments: statutes and laws aimed to set forth the policy statements and principles, as well as the responsibilities for protection and safety at national (federal), state and institutional levels.



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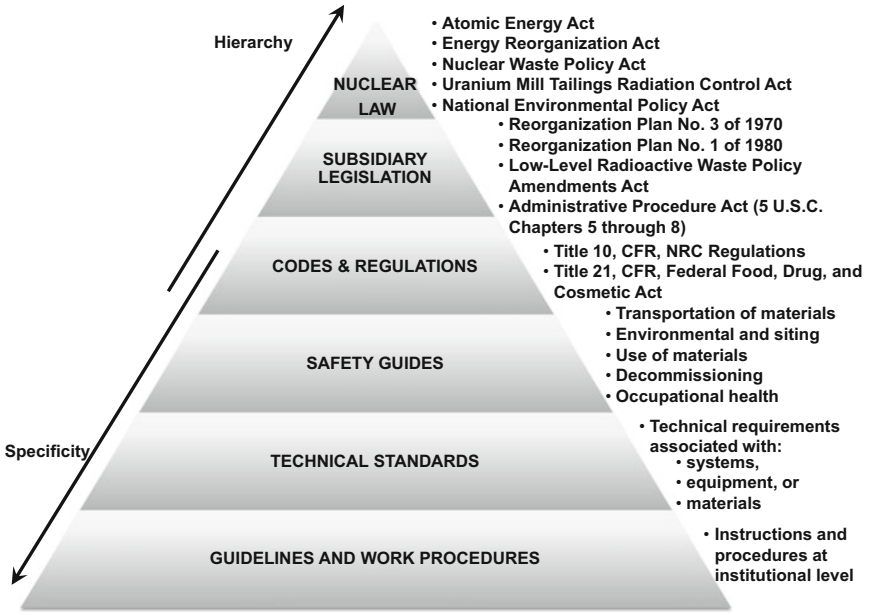


**Fig. 9.1** Legal framework hierarchy

The following groups include regulations to implement and administer the requirements and the effective coordination between participants, especially where regulatory responsibilities for radiation safety are divided.

More specific regulations, comprising administrative and technical requirements, are to be amended frequently, as knowledge is gained from scientific and technical developments, though the more specific the regulation—standards, guidelines, etc.—the easier amending they should be. Specific guidelines and work and safety procedures at institutional level are also included in these groups.

Governments are required to establish a national authority through law to regulate and control the introduction and conduct of any practice involving radiation sources, including the requirements governing notification and authorization, and issuing authorizations itself, thus, thereby regulating and enforcing nuclear, radiation, radioactive waste, and transport safety [1]. The responsibility at national level for different exposure situations, or different aspects of radiation protection and safety, may be divided between different authorities.

The regulatory authority[1] is also empowered by the government to issue detailed regulatory instruments in matters requiring its expertise, including specific regulations emphasizing and supporting its primary responsibility in achieving and

[1]The regulatory authority is an entity or a system of authorities designated by the government as having legal authority for conducting the regulatory process, and thereby regulating nuclear, radiation, radioactive waste and transport safety.

maintaining a satisfactory control. For example, the regulatory authority establishes and implements an enforcement policy within the legal framework for responding to noncompliance with regulatory requirements or with any conditions specified in the authorizations [2].

To ensure that the judgments and decisions of the regulatory authority are not influenced by any interests other than safety, the same legislation by which the regulatory authority is created should state its completely independence from any government departments and agencies, or organizations, conducting or promoting activities where radiation sources are involved. In addition, the legislation should also make clear that the regulatory authority is independent of registrants, licensees, and the designers and constructors of the radiation sources.

It is recommended that regulatory authorities make the arrangements with other entities for performing essential activities and providing services—calibration, training, or dosimetric services—that are beyond the capabilities of operators and which are not otherwise available, but do not provide these services [1].

The management system provides for external factors, interrelated or interacting with the safety objectives, to be identified—specific ministries; health, labor, occupational, and environmental authorities; manufacturers, providers, exporters and importers; customs and border protection, and homeland security organizations; and different specialized institutions and societies, etc.—to ensure that health, environmental, security, quality, and economic requirements are considered to improve the safety performance in normal, transient, and emergency situations [4].

Governments through law also establish the operator prime responsibility for safety. The operator is responsible for ensuring safety in the siting, design, construction, commissioning, operation, decommissioning, closeout, or closure of its facilities, including, as appropriate, rehabilitation of contaminated areas; and for activities in which radioactive materials are used, transported or handled. Organizations which generate radioactive waste shall have responsibility for the safe management of the radioactive waste they produce [2].

A minimum of regulatory instruments should at least be part of the regulatory framework:

(a) The principles and limits acknowledged by the International Basic Safety Standards for Radiation Protection and Safety of Radiation Sources [3].
(b) Provisions for notification, registry, and licensing of radiation sources.
(c) Provisions for inspection and assessment of compliance with regulations.
(d) Enforcement actions for responding to noncompliance by authorized parties with regulatory requirements or with any conditions specified in the authorization.
(e) Criteria for exemption of specific sources from regulatory control.
(f) Methods to deal with possible accidents and for emergency preparedness.
(g) Procedures to report and release information about safety and protection.

## 9.2 Regulatory Control

The regulatory control main objective is to protect individuals, society as a whole, and the environment against the harmful effects of ionizing radiation. A regulatory control is established upon a well-structured system of notification, authorization, and inspection supported by the regulatory authority and enforcement actions.

Regulatory control[2] [5] applies to any act of possessing, using, processing, exporting, importing, transporting, distributing, and disposing of radioactive materials and radiation generating machines or devices, except when there is no need of such application, e.g., the transport or disposal of electronic devices that only emit radiation when connected to power. Furthermore, regulatory control applies to the design, construction, operation, and decommissioning of equipment and facilities that possess, use or process radiation sources, referring by them whatever may cause radiation exposure. Regulatory control also applies to the management of radioactive wastes, and to the siting, design, construction, operation, and closure of sites used for the disposal of such wastes.

Likewise, radiological regulatory control applies to occupational and public exposures, including potential exposures, associated with any planned, emergency or existing exposure situations, and to medical exposures only for planned exposure situations.

The regulatory control should also identify and exclude exposures that are not amenable to control. Excluded exposures are those that either cannot be restricted by a regulatory action or for which control is obviously impracticable. Typically, these are exposures from $^{40}K$ in the body, from cosmic radiation at the surface of the earth and from naturally occurring radioactive materials (NORM), in which the activity concentrations of natural radionuclides are below the relevant values given for exclusion [6]. In addition, these regulations should ensure exemption, from all or some regulatory requirements, of sources or situations that are unwarranted to be controlled on the basis that the exposure, including potential exposure, is judged to be too small to warrant the application of those aspects [7].

## 9.3 Purpose and Scope of Specific Regulations

Criteria, upon which radiological requirements are based, have been continuously developing with the increase of scientific knowledge and technology advancement. For the first 60 years after the discovery of ionizing radiation, the purpose of radiological protection was that of avoiding deterministic effects from occupational exposures, and to keep individuals below the relevant thresholds. The concept of critical organ was introduced in the 1954 ICRP Recommendations, and the

[2]Any form of control or regulation applied to facilities or activities by a regulatory authority for reasons relating to radiation protection or to the safety or security of radioactive sources.

recommended dose limit was related to the organs that were said to be critical in the case of whole body exposure, i.e., the gonads and the blood-forming organs. In the 1960s, the ICRP summarized for the first time the current knowledge about radiation risks, both somatic and genetic [8]. The IAEA issued its first basic safety standards for protecting workers and the public against excessive radiation in 1962, the Health and Safety Measures (INFCIRC/18).

Publication 26 (ICRP 1977) [9] set out the new system of dose limitation and introduced the three principles of protection; the IAEA basic standards were correspondingly reviewed in 1982. In 1990, when the ICRP published a new set of recommendations [10], the international and regional agencies concerned—World Health Organization (WHO), International Labor Organization (ILO), Food and Agriculture Organization (FAO), Nuclear Energy Agency (NEA), and Pan American Health Organization (PAHO)—as well as the IAEA, reviewed the IAEA's standards of 1982 [11] and issued the international basic standards in 1996 [12]. A recent revision of these standards was completed in 2014 with the publication of GSR Part 3 [3] following 2007 ICRP Recommendations [13].

Considering the scope, contents, and level of detail, specific regulations in radiological protection should consist of both descriptive and prescriptive regulations.

Descriptive regulations, also known as performance regulations, promote uniformity, equality and a baseline of protection. They include more comprehensive requirements that are easier to prepare and do not need so many frequent changes. This type of regulation is used to establish general safety principles and requirements, and basic operation parameters [1]. For example, items like notification requirements for licensing, safety, and security principles, general requirements applicable to workers, public, and patients, administrative procedures for exemption of sources, and so on, can be subject to descriptive regulations.

Prescriptive regulations are best used in technical standards, guidelines, specific rules, and communications, etc. They are goal-based and take into consideration the best engineering practice of achieving compliance; hence they need to be updated when technologies advance. Prescriptive regulations state how to achieve radiation safety—what techniques or instruments to use, what qualifications are needed, where and how specific functions are be performed [1]. This type of regulation can be used for establishing: inspection procedures; safety provisions for the design of specific facilities; detailed requirements for safety, etc. Prescriptive regulations require less time and skills to perform, e.g., an inspection, but they demand more knowledge and expertise, as well as more frequent modifications.

The development of any particular radiation safety regulation will involve a balance between two concerns—the need for flexibility to permit easy adaptation of the regulations to evolving circumstances and technology versus the need to include detailed requirements for safety.

Figure 9.2 represents such balance. There are two levels of specific regulations, i.e., a set of descriptive regulations, including codes, guidelines, standards, and other provisions usually emitted by the regulatory authority or specific technical, scientific and/or governmental agencies, and a set of corresponding instructions and

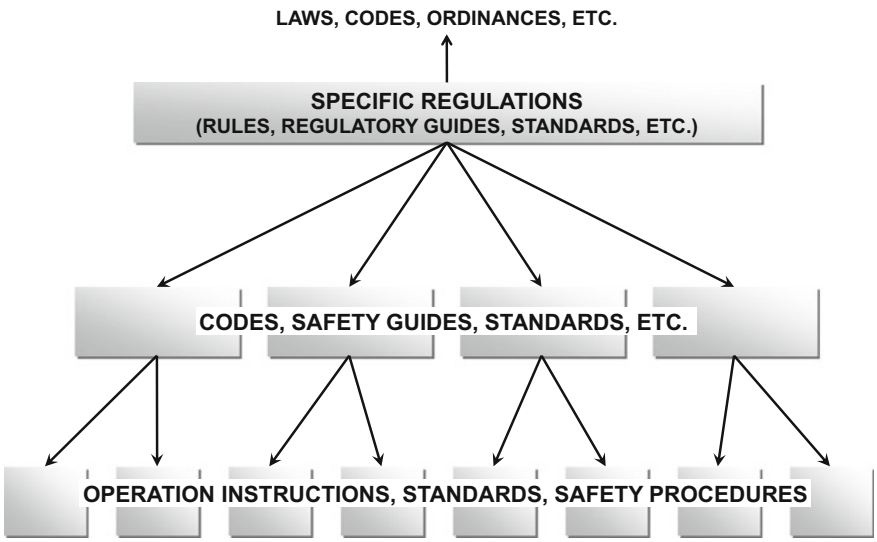


**Fig. 9.2** Balance between descriptive and prescriptive regulations

procedures explaining how to accomplish those requirements. These can be prescriptive regulations emitted by the regulatory authority, specific agencies, or local authorities, etc.

For example, Part 35 of Title 10, Code of Federal Regulations, establishes the requirements and provisions for the medical use of byproduct material and for the issuance of specific licenses authorizing the medical use of this material [14]. This is a descriptive regulation. Then, by using regulatory guides, the Nuclear Regulatory Commission (NRC) provides guidance to licensees, applicants, and stakeholders on, e.g., the leak testing of radioactive brachytherapy sources or the verification of containment properties of sealed radioactive sources [15]. Both last examples are prescriptive regulations.

The NRC also undertakes a variety of activities to integrate risk information and performance measures into the agency's regulations, regulatory guidance, and oversight processes [16]. Using risk information helps to reduce unnecessary requirements in design and operation of facilities that, otherwise, could be overestimated, while a performance-based approach, focused on the results as the primary basis for regulatory decision-making, can be used to identify a wide range of options to improve safety in a given facility.

Standards may be published documents establishing a model for technical specifications and recommended practices for performance or safety. Just to give an idea of the broad scope of these standards, they can cover from the symbol of ionizing radiation, important definitions, limits, and rules to structural shielding designs and management procedures for specific applications.

Standards can be enforced by law, e.g., 10 CFR Part 20, Standards for Protection against Radiation [17], or can be proprietary—controlled by one company—or developed by standards organizations by consultation and consensus.

A standards organization could be a governmental, quasi-governmental, or nongovernmental entity whose primary activities are the development and maintenance of standards. Typically, standards organization works by technical experts committees from facilities, vendors, and governmental agencies. It represents itself and/or the government at international standards organizations like the International Organization for Standardization (ISO) [18] and the International Electrotechnical Commission (IEC) [19]. Regulatory authorities might coordinate with standards organizations to develop specific standards, and use existing standards to improve their effectiveness and reduce unnecessary efforts.

## 9.4 Examples of Regulatory Control in United States

The Atomic Energy Act of 1954[3] is the fundamental law on both the civilian and the military uses of nuclear materials in the United States. The Act requires that civilian uses of nuclear materials and facilities be licensed, and it empowers the Nuclear Regulatory Commission (NRC) to establish by rule or order, and to enforce, such standards governing the uses as "the Commission[4] may deem necessary or desirable in order to protect health and safety and minimize danger to life or property [20]." By the Energy Reorganization Act of 1974, regulatory and developing functions of the preceding Atomic Energy Commission were set apart, and the Nuclear Regulatory Commission was fully established as an independent regulatory authority for licensing and related regulatory tasks pursuant to chapters referred to special material, source material, byproduct material, and atomic energy licenses of the Atomic Energy Act of 1954. Federal research and development work for all energy sources, as well as nuclear weapons production, is now conducted by the U.S. Department of Energy.

Under Sect. 274 of the Atomic Energy Act of 1954, the NRC may enter into an agreement with a State for discontinuance of the NRC's regulatory authority over some materials licensees within the State. The State must first show that its regulatory program is compatible with the NRC's and adequate to protect public health and safety. The NRC retains authority over, among other things, nuclear power plants within the State and exports from the State [20].

NRC licenses the following activities: (a) construction, operation, and decommissioning of commercial reactors and fuel cycle facilities; (b) possession, use, processing, exporting, and importing of nuclear materials and waste, and certain

[3]All references to the Atomic Energy Act of 1954 and to the Energy Reorganization Act of 1974 are "as amended".

[4]The former Atomic Energy Commission.

aspects of its transportation; (c) siting, design, construction, operation, and closure of waste disposal sites. The licensing process includes approving the initial license, subsequent license modifications, and license renewals.

The inspection program of nuclear facilities in the United States is carried out by the NRC's regional offices located in King of Prussia, Pennsylvania (Region I, Northeastern U.S.); Atlanta, Georgia (Region II, Southeastern U.S.); Lisle, Illinois (Region III, Midwestern U.S.); and Arlington, Texas (Region IV, Western and southern Midwestern U.S.). In addition to region-based inspectors, the NRC also has resident inspectors—on-site—stationed at each nuclear facility; they oversight activities at the plants and check on adherence to federal safety requirements on a daily basis.

Pursuant subsection 274b of the Atomic Energy Act, the NRC has entered into an effective regulatory discontinuance agreement with 37 States to regulate most of radioactive materials. They include source material (uranium and thorium), reactor fission byproducts, and quantities of special nuclear materials (SNM) not enough to form a critical mass. Under its own internal practices, the NRC periodically reviews the performance of each Agreement State to assure compatibility with NRC's regulatory standards.

Pursuant to the Radiation Control for Health and Safety Act of 1968 (now Subchapter C of the Electronic Product Radiation Control of the Federal Food, Drug, and Cosmetic Act (revised/posted 1-23-99), the Food and Drug Administration (FDA) oversees radiation safety of medical imaging systems, counter-terrorism security systems (such as baggage and passenger screening systems for airline security), and industrial and electronic consumer products that emit radiation. The FDA's functions regarding these products include setting standards, recommending good practices, conducting research, and educating manufacturers and consumers. The Nuclear Regulatory Commission shares authority with the FDA regulating radiation safety in Agreement States. For example, the FDA regulates the manufacture and use of linear accelerators in the production of radioisotopes for nuclear medicine, but the States regulate the operation of such devices.

Along with the registry and control of devices generating ionizing radiation, compliant with FDA regulations, Agreement States issue radioactive material licenses, enact their own regulations based on the federal code, and enforce those regulations under the authority of each individual state's laws. Each state inspects their own facilities, e.g., hospitals and universities that use radioactive materials and radiation emitting equipment (*X*-ray, radiographic and fluoroscopic machines, etc.); certifies those who operate such machines or manage radioactive materials; and provide Emergency Preparedness and Emergency Response capabilities. Agreement States exercise their licensing and enforcement actions under the authority of the governors [21].

The Organization of Agreement States (OAS) is a nonprofit, voluntary, scientific, and professional society, integrated by the state radiation control directors and staff from the 37 Agreement States. The purpose of this organization is to work together and with the NRC on regulatory issues associated with the respective agreements [22].

Furthermore, each state in United States, Agreement or non-agreement, has one or more programs designed to assure and assess the proper use of radiation sources, and its own radiological monitoring capabilities for environmental protection and emergency preparedness. Radiation FDA's undertakings, and NRC's regulatory tasks are very close related to public health and environmental matters at state level.

The Conference of Radiation Control Program Directors, Inc. (CRCPD, is a nonprofit nongovernmental professional organization, dedicated to radiation protection. CRCPD's primary membership is comprised of radiation professionals in State and local government that regulate the use of radiation sources [23]. CRCPD is a network of cooperation, enforcement programs, exchange of information, and regulation harmonization. It also provides training and technical assistance. The Suggested State Regulations for Control of Radiation (SSRCRs) is a CRCPD Dynamic Document that is revised and updated on an ongoing basis. It is a comprehensive regulatory framework for the States covering all radiation sources and activities—radioactive materials, licensing, registration, inspection, management of low-level radioactive waste, *X*-ray, accelerators, emergency response, decommissioning, environmental monitoring, radon, and security, etc. SSRCRs are supported financially by the US Food and Drug Administration Cooperative Agreement.

The CRCPD also offers assistance in finding affordable, legal disposition for unwanted or orphan radioactive sources, in responding when radioactivity is detected in metal and scrap, as well as guidance for the disposition of contaminated scrap, and the use of detectors at landfills [24].

CRCPD has formal working relationships with federal agencies and organizations that either regulate radiation sources or are involved in radiation protection, such as:

- Nuclear Regulatory Commission (NRC)
- Food & Drug Administration (FDA)
- Environmental Protection Agency (EPA)
- Federal Emergency Management Agency (FEMA)
- Department of Energy (DOE)
- Department of Homeland Security
- Department of Transportation (DOT)
- Centers for Disease Control (CDC)
- National Institute of Standards (NIST)
- National Academy of Sciences (NAS)
- National Institute of Occupational Health and Safety (NIOSH)

By the Reorganization Plan No. 3 of 1970, the U.S. Environmental Protection Agency (EPA) was empowered to establish "generally applicable environmental standards for the protection of the general environment from radioactive material." Also, the Reorganization Plan No. 1 of 1980 strengthened the executive and administrative roles of the NRC in emergencies, transferring to the NRC's Chairman "all the functions vested in the Commission pertaining to an emergency concerning a particular facility or materials (…) regulated by the Commission." [20].

Organizations that produce standards in the United States—they are many—normally are accredited by the American National Standards Institute (ANSI). The National Council on Radiation Protection and Measurements (NCRP) is a U.S. organization that seeks to formulate and disseminate information, guidance, and recommendations on radiation protection and measurement [25]. The Radiation Physics Division, part of the Physical Measurement Laboratory at the National Institute of Standards and Technology (NIST) also develops, maintains, and disseminates the national standards for ionizing radiation and radioactivity [26].

At institutional level, registrants, and licensees, i.e., power plants, fuel cycle facilities, academic and research institutions, hospitals, clinics, radiographic and industrial enterprises, radioactive waste disposal facilities, etc., are required to implement a management system for safety consistent with the type and magnitude of the sources they own and the risk of exposure. All institutional management systems must define the organization and main responsibilities of all levels to safety; and establish the necessary operational and safety instruction, rules, and procedures as stated by international, national, and state requirements to ensure radiological protection of workers and member of the public.

## References

1. International Atomic Energy (2004) Regulatory control of radiation sources. Safety Guide No. GS-G-1.5 Jointly sponsored by FAO, ILO, PAHO and WHO. IAEA, Vienna
2. International Atomic Energy (2010) Governmental, legal and regulatory framework for safety, Safety Standards No. GSR Part 1. IAEA, Vienna
3. International Atomic Energy (2014) Radiation protection and safety of radiation sources: International Basic Safety Standards, IAEA Safety Standards Series GSR Part 3. IAEA, Vienna
4. International Atomic Energy (2006) The management system for facilities and activities, Safety Requirements No. GS-R-3. IAEA, Vienna
5. International Atomic Energy (2004) Code of conduct on the safety and security of radioactive sources. IAEA, Vienna
6. International Atomic Energy (2004) Application of the concepts of exclusion, exemption and clearance, Safety Standards Series No. RS-G-1.7. IAEA, Vienna
7. International Atomic Energy (2007) IAEA safety glossary, terminology used in nuclear safety and radiation protection—2007 Edition. IAEA, Vienna
8. ICRP Publication 109 (2009) Application of the commission's recommendations for the protection of people in emergency exposure situations. Ann ICRP 39(1)
9. ICRP Publication 26 (1977) Recommendations of the ICRP. Ann ICRP 1(3)
10. ICRP Publication 60 (1991) 1990 recommendations of the international commission on radiological protection. Ann ICRP 21(1–3) (1991)
11. Fischer D (1997) HISTORY of the international atomic energy agency, the first forty years, STI/PUB/1032. IAEA, Vienna
12. IAEA, FAO, ILO, OCDE/NEA, PAHO, WHO (1996) International basic safety standards for protection against ionizing radiation, Safety Series No. 115. IAEA, Vienna
13. ICRP Publication 103 (2007) The 2007 recommendations of the international commission on radiological protection. Ann ICRP 37(2.4)

14. U.S. NRC (2015) NRC Regulations, Title 10, Code of Federal Regulations, United States Nuclear Regulatory Commission. http://www.nrc.gov/reading-rm/doc-collections/cfr/
15. U.S. NRC (2015) Products (Division 6): Regulatory Guides 6.1−6.9. United States Nuclear Regulatory Commission. http://www.nrc.gov/reading-rm/doc-collections/reg-guides/products/rg/division-6/division-6-1.html
16. U.S. NRC (2015) Risk assessment in regulation. United States Nuclear Regulatory Commission. http://www.nrc.gov/about-nrc/regulatory/risk-informed.html
17. U.S. NRC (2015) 10 CFR PART 20—Standards for protection against radiation. United States Nuclear Regulatory Commission. http://www.nrc.gov/reading-rm/doc-collections/cfr/part020/
18. ISO Standards (2015) International Organization for Standardization. http://www.iso.org/iso/home/standards.htm
19. Standards development (2015) International Electrotechnical Commission (IEC). http://www.iec.ch/standardsdev/?ref=menu
20. NRC, Statutory Authority. USNRC. http://www.nrc.gov/about-nrc/governing-laws.html
21. U.S. NRC, Regulation of radioactive materials. United States Nuclear Regulatory Commission. http://www.nrc.gov/about-nrc/radiation/protects-you/reg-matls.html
22. Organization of Agreement States (2015) Organization of Agreement States, OAS, 2015. http://www.agreementstates.org/
23. Conference of Radiation Control Program Directors (2015) A partnership dedicated to radiation protection. Conference of Radiation Control Program Directors, Inc. http://www.crcpd.org/
24. O'Kelley TP (2007) A model for networking, cooperation, resource sharing, information exchange and regulation, IAEA-CN-142/54. In: International Conference on the challenges faced by technical and scientific support organizations. Aix-en-Provence
25. NCRP (2015) national council on radiation protection & measurements. National Council on Radiation Protection & Measurements. http://www.ncrponline.org/
26. NIST (2015) Physical Measurement Laboratory, Radiation Physics Division. The National Institute of Standards and Technology (NIST). http://www.nist.gov/pml/div682/index.cfm

# Chapter 10
# The Management System for Safety

Aside from the basic elements of time, distance and shielding, and a comprehensive set of regulations and high-quality standards, a management system properly established, and a solid groundwork for notification, authorization, and inspection are key components for a stable and sustainable safety regime.

The basic requirements for the protection of people and environment from harmful effects of ionizing radiation and for the safety of radiation sources are stated in the International Basic Safety Standards [1] and expected to be reflected in the corresponding ruling at national level. Such requirements take into consideration the most recent findings of United Nations Scientific Committee on the Effects of Atomic Radiation (UNSCEAR) [2] and also the latest recommendations of the International Commission on Radiological Protection (ICRP) [3].

The extent to which the requirements are applied depends on the nature and magnitude of the sources, and the type, dose, and likelihood of exposures. In any case, to build a strong safety culture, the management system is based on the following safety principles [4]:

- Facilities and activities that give rise to radiation risks must yield an overall benefit, i.e., must do more good than harm.
- The prime responsibility for safety rests with the person or organization responsible for facilities and activities that give rise to radiation risks.
- Facilities and activities are in compliance with the regulatory framework, including established standards.
- Facilities and activities have an effective leadership and management system for safety demonstrated at the highest levels in the organization.
- Protection is optimized to provide the highest level of safety that can reasonably be achieved.
- Dose is limited so that no individual bears an unacceptable risk of harm, i.e., measures are taken to ensure that the specified dose limits for occupational exposure and those for public exposure are not exceeded.



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- People and environment of present and future generations are protected against radiation risks.
- All practical efforts are in place to prevent and mitigate nuclear or radiation accidents.
- Arrangements are made and maintained for emergency preparedness and response for nuclear or radiation incidents.
- Protective actions to reduce existing or unregulated radiation risks are justified and optimized.

A management system integrates the interrelated or interacting elements for safety—health, environmental, security, quality, and economic—that establish policies and objectives, and enables those objectives to be achieved in a safe, efficient, and effective manner. The main objectives of such management system are [5]

1. To bring together in a coherent manner all the requirements;
2. To describe the planned and systematic actions necessary to provide adequate confidence that all these requirements are satisfied; and
3. To ensure that health, environmental, security, quality, and economic requirements are not considered separately from safety requirements.

There are also two specific aims of the management system at the level of facilities and activities

- To improve the safety performance of the organization through the planning, control, and supervision of safety-related activities in normal, transient, and emergency situations;
- To foster and support a strong safety culture through the development and reinforcement of good safety attitudes and behavior in individuals and teams, so as to allow them to carry out their tasks safely.

## 10.1 Interested Parties to the System

Interested parties may include: “customers, owners, operators, employees, suppliers, partners, trade unions; the regulated industry or professionals; scientific bodies; governmental agencies and/or regulatory authorities whose responsibilities may cover nuclear energy; the media; the public—individuals, community groups and interest groups—; and other governments, especially those that have entered into agreements providing for an exchange of information concerning possible transboundary impacts, or those involved in the import or export of certain technologies or materials” [6] [“Reproduced with permission by the IAEA”].

### 10.1.1 The Regulatory Authority

The regulatory authority[1] is responsible for the establishment of the requirements and guidelines to protect the people and the environment in all exposure situations; for the authorization, control and inspection of facilities and activities; and for the enforcement of legislative and regulatory provisions.

To accomplish its responsibilities, the regulatory authority establishes and puts into practice a comprehensive system for notification and authorization, and regularly conducts a systematic assessment of the hazards associated with facilities and activities by inspection and reporting. Persons and organizations are required to notify and apply for authorization and to attach all the supporting information to demonstrate that the projected facility or activity is safe enough according to the corresponding standards.

Unless the source is excluded or exempted from regulatory control, the regulatory authority is expected to grant such authorization in the form of a registry or a license after processing all the received information. The regulatory authority also requires from facilities and activities the adequate level of personnel training, qualification, and competence, as well as the establishing of dose constraints or reference levels, as appropriate, for the optimization of safety and protection.

The regulatory authority usually creates and maintains general registries of authorizations and occupational doses, as well as inventories of sources and radioactive waste. It also provides for the control of sources for which no other organization has responsibility; conducts and requests investigations for all abnormal occurrences and timely disseminates information on radiation safety.

### 10.1.2 Responsibility and Authority

The management system clearly establishes the lines of responsibility and authority for safety. Responsibility implies being accountable for—or having obligations or duties in regard to—safety and protection at all levels of the organization, including being in compliance with all established rules and procedures. Responsibility also entails the application of effective relationships and channels of communication. Authority implies to use and allocate the resources efficiently, and to make decisions to achieve the objectives effectively. For facilities and activities to perform safely, responsibility and authority should come along, combined in a reasonable manner.

The top management of facilities and activities—a power plant, hospital, clinic, academic or research institution, radiographic service, public or private waste storage facility, source manufacturer or supplier, etc., bears the prime responsibility for radiation safety. It is also responsible for providing the means and resources to other levels with responsibility—middle managers, first-line managers, radiation

[1]Bear in mind that regulatory authority could be a system of authorities like in the United States.

safety officers, experts, and workers—for them to achieve their specific objectives for protection and safety.

Radiological medical practitioners[2] bear responsibility for patient protection and safety in the planning and delivery of medical exposures. They should have authority to assure that no patient receives a medical exposure unless it has been justified in consultation with the referring medical practitioner, as appropriate, or approved by an ethics committee, if the exposure is part of an approved biomedical research program. Medical physicists, medical radiation technologists, radiopharmacists, and other health professionals with specific duties are accountable for the appropriate techniques, procedures and software, calibration, dosimetry and quality assurance, etc.

In emergency or existing exposures, the prime responsibility relies on those who have to deal with the situation—emergency response teams, national or local authorities, specialized agencies, etc. Emergency workers, workers undertaking remedial works, relevant authorities, etc., also bear their own responsibilities for safety and protection.

Accountabilities for safety are to be specifically described—of the managers of the different facilities and activities in relation to occupational exposure; of the radiological medical practitioners in relation to medical exposure, and of the designated persons who undertake the leading role in emergency exposure situations or existing exposure situations. Also the responsibilities of workers; radiation protection officers; referring medical practitioners; medical physicists; medical radiation technologists; other qualified experts; ethical review committees; and of any other to whom the administration has delegated a specific responsibility for safety.

Protection and safety are to be "effectively integrated into the existing overall management system" [1] ["Reproduced with permission by the IAEA"], including the quality assurance system, so that: they are not compromised by other requirements or demands; all required notifications and authorizations are up-to-date; and a regular assessment of safety performance and the application of lessons learned from experience are warranted.

## 10.2 Notification and Authorization

"Any person or organization intending to operate a facility, or to conduct an activity, is required to first submit to the regulatory authority a notification and an application for authorization, as appropriate" [1] ["Reproduced with permission by the IAEA"]. Notification and authorization are usually scaled consistent with the level of risk, the complexity of the sources and operations, and the likelihood of

[2]A health professional with specialist education and training in the medical uses of radiation, who is competent to perform independently or to oversee a radiological procedure—a procedure in diagnostic radiology, nuclear medicine or radiation therapy, or a planning procedure, image guided interventional procedure or other interventional procedure involving radiation.

exposures. This requirement is crucial to set priorities on the details that persons and organizations should fulfill in regard to safety assessments and the overall measures they should implement, as well as the efforts the regulatory authority has to develop to supervise the safety and security of such facilities and activities.

### *10.2.1 Notification*

Although notification is a common procedure for reporting any event associated with a possible exposure—an emergency or nonemergency incident, a nuclear plant shutdown, a shipment of radioactive material, a purchase of a new equipment or source, and so on—it is also the lower level of authorization.

Notification is generally required to operate a facility or to conduct an activity. But, since issues of safety and security may arise during each stage of the life of a source, notification is also required for the design, manufacture, construction, assembly, installation, disassembly, acquisition, import, export, and distribution of a source, or apparatus containing a radiation source, as well as for its transfer, transportation, repair, storage, and disposal. It is required for the manufacture, assembly, maintenance, import, and distribution of consumer products containing radioactive materials too. Each regulatory authority has its own guidelines and procedures for notifications.

Notification alone is sufficient for sources from which normal exposures are expected to be very small, but that are not suitable for exemption for some reason; or from which potential exposures are considered negligible and the practice does not imply an unwarranted hazard to public health and safety [7]. On a case by case basis, it is also possible to provide an exemption following notification.

Some sources or activities are exempted by the rules. For example, uranium contained in counterweights installed in aircraft are exempted if they were manufactured in accordance with a specific license issued by the NRC authorizing their distribution [8]; electronic equipment that produces radiation, incidental to its operation for other purposes, is exempt from registration and notification requirements if the dose equivalent rate averaged over an area of ten square centimeters does not exceed 5 μSv per hour at five centimeters from any accessible surface of the equipment [9].

### *10.2.2 Authorization*

Authorization is “a written permission granted by the regulatory authority to perform specified activities, after a safety analysis evaluation and/or inspection” [6] [“Reproduced with permission by the IAEA”]. An authorization can take the form of a license, certification, or registration. Registration is used to authorize practices of low or moderate risks, while licenses are used for facilities, activities, and

**Table 10.1** Categories for sealed sources used in common practices ["Reproduced from [10] with permission by the IAEA"]

| Category | Radionuclide source, equipment or machine | Activity ratio (A/D) |
|---|---|---|
| 1 | Radioisotope thermoelectric generators (RTGs)<br>Irradiators<br>Teletherapy sources<br>Fixed, multi-beam teletherapy (gamma knife) sources | $A/D \geq 1000$ |
| 2 | Industrial gamma radiography sources<br>High/medium dose rate brachytherapy sources | $1000 > A/D \geq 10$ |
| 3 | Fixed industrial gauges that incorporate high activity sources<br>Well logging gauges | $10 > A/D \geq 1$ |
| 4 | Low dose rate brachytherapy sources (except eye plaques and permanent implant)<br>Industrial gauges that do not incorporate high activity sources<br>Bone densitometers<br>Static eliminators | $1 > A/D \geq 0.01$ |
| 5 | Low dose rate brachytherapy eye plaques and permanent implant sources<br>*X*-ray fluorescence (XRF) devices<br>Electron capture devices<br>Mossbauer spectrometry sources<br>Positron Emission Tomography (PET) check sources | $0.01 > A/D$ and<br>$A >$ exempt |

materials with higher level of risk. Certification is usually used to authorize the source or device itself—for example, as a sealed source.

Facilities and activities that are typically amenable to registration are those for which [1] ["Reproduced with permission by the IAEA"]:

- "Safety can largely be ensured by the design of the facilities and equipment;
- Operating procedures are simple to follow;
- Safety training requirements are minimal; and
- There is a history of few problems with safety during operation."

The type of authorization is generally consistent with the categorization of sources. Table 10.1 shows the categories for sealed sources used in common practices, recommended by the IAEA [1, 10]. This categorization provides a relative ranking in terms of a D value, i.e., the activity corresponding to a dangerous source. D value is defined as "the specific activity of a source which, if not under control, could cause severe deterministic effects for a range of scenarios that include both external exposure from an unshielded source, and internal exposure following dispersal of the source material" [10]. D values from the International Safety Standards are shown in Table 10.2 [1]. D values for additional radionuclides and full details of scenarios and exposure pathways considered in determining the D values are provided in IAEA EPR-D-Values [11].

**Table 10.2** Activity corresponding to a dangerous source (D value) for selected radionuclides ["Reproduced from [1] with permission by the IAEA"]

| Radionuclide D value | (TBq) | Radionuclide D value | (TBq) |
|---|---|---|---|
| $^{241}$Am | $6 \times 10^{-2}$ | $^{99}$Mo | $3 \times 10^{-1}$ |
| $^{241}$Am-/Be | $6 \times 10^{-2}$ | $^{63}$Ni | $6 \times 10^{1}$ |
| $^{198}$Au | $2 \times 10^{-1}$ | $^{32}$P | $1 \times 10^{1}$ |
| $^{109}$Cd | $2 \times 10^{1}$ | $^{103}$Pd | $9 \times 10^{1}$ |
| $^{252}$Cf | $2 \times 10^{-2}$ | $^{147}$Pm | $4 \times 10^{1}$ |
| $^{244}$Cm | $5 \times 10^{-2}$ | $^{210}$Po | $6 \times 10^{-2}$ |
| $^{57}$Co | $7 \times 10^{-1}$ | $^{238}$Pu | $6 \times 10^{-2}$ |
| $^{60}$Co | $3 \times 10^{-2}$ | $^{239}$Pu-/Be | $6 \times 10^{-2}$ |
| $^{137}$Cs | $1 \times 10^{-1}$ | $^{226}$Ra | $4 \times 10^{-2}$ |
| $^{55}$Fe | $8 \times 10^{2}$ | $^{106}$Ru ($^{106}$Rh) | $3 \times 10^{-1}$ |
| $^{153}$Gd | $1 \times 10^{0}$ | $^{75}$Se- | $2 \times 10^{-1}$ |
| $^{68}$Ge | $7 \times 10^{-2}$ | $^{90}$Sr ($^{90}$Y) | $1 \times 10^{0}$ |
| $^{3}$H | $2 \times 10^{3}$ | $^{99m}$Tc | $7 \times 10^{-1}$ |
| $^{125}$I | $2 \times 10^{-1}$ | $^{204}$Tl | $2 \times 10^{1}$ |
| $^{131}$I | $2 \times 10^{-1}$ | $^{170}$Tm | $2 \times 10^{1}$ |
| $^{192}$Ir | $8 \times 10^{-2}$ | $^{169}$Yb | $3 \times 10^{-1}$ |
| $^{85}$Kr | $3 \times 10^{1}$ | | |

The exposure scenarios taken from the experience of real accidents in industrial radiography and/or from orphan sources that were evaluated to determine the D value are

(a) An unshielded source being carried in the hand for one hour, in a pocket for 10 h, or being in a room for days to weeks, or
(b) Dispersal of a source, e.g., by fire, explosion, or human action, resulting in a dose from inhalation, ingestion, and/or skin contamination.

The A value represents the actual activity of the source. The A/D value is used to provide an initial ranking of relative risk.

Dose criteria for internal exposure are: 1 Gy to the bone marrow or 6 Gy to the lung from low LET radiation received by the organ in 2 days; 25 Gy to the lung from inhaled radionuclides of high LET radiation in 1 year; and 5 Gy to the thyroid received by the organ in 2 days.

For external exposure, dose criteria are: for a source in contact with tissue, more than 25 Gy—the threshold for necrosis—at a depth of 2 cm for most parts of the body (e.g., from a source in a pocket) or 1 cm for the hand; and for a source that is considered too big to be carried, 1 Gy to the bone marrow in 100 h at a distance of 1 m.

Sources of fourth and fifth categories—thickness gauges or fill-level thickness gauges, ophthalmic devices, static eliminators or electronic capture detectors, and small calibration sources, etc.—are usually suitable for registration. In many countries the manufacture of such devices is authorized by license; and the user of the same device is authorized by notification or registration.

### 10.2.3 Licensing in the United States

Licenses in the United States are of two types: general and specific. A general license is issued by NRC, or by Agreement States, to persons or organizations which acquire, receive through an authorized transfer, possess, use or transfer the following [12]:

- Certain devices and equipment containing a total of < 18.5 MBq of $^{210}$Po per device or < 1850 MBq of $^{3}$H per device;
- Certain detecting, measuring, gauging, or controlling devices, and certain devices for producing light or an ionized atmosphere (e.g., gas chromatograph units; moisture/density, fill level, insertion and transmission gauges; static eliminators, self-luminous exit signs, and ion generating tubes). State regulations require that general licensees who possess devices containing at least 370 MBq of $^{137}$Cs, 3.7 MBq of $^{90}$Sr, 37 MBq of $^{60}$Co, or 37 MBq of $^{241}$Am or any other transuranic (i.e., element with atomic number greater than uranium) to register with the NRC, to increase their control and accountability, and to prevent them from becoming orphan sources;
- Luminous safety devices for use in aircraft: specifically $^{3}$H or $^{147}$Pm contained in luminous safety devices, such as luminous exit signs and dials, for use in aircraft, except that $^{147}$Pm is not generally licensed in instrument dials. Devices may contain no more than 370 GBq of $^{3}$H or 11.1 GBq of $^{147}$Pm;
- $^{241}$Am and $^{226}$Ra in the form of calibration or reference sources; 185 kBq at any one time in a location of use or storage of such sources;
- General license for $^{90}$Sr in ice detection devices provided that the devices contain no more than 1850 kBq.

A general license allows receiving and using the above devices and sources, only if it has been manufactured and distributed in accordance with a specific license issued by the NRC or by an Agreement State. The general licensee must comply with the requirements for labeling, instructions for use, and proper storage or disposition of the device.

As stated by 10 CFR 31.5 [13], a device containing radioactive material typically used to detect, measure, gauge, or control thickness, density, level, or chemical composition, is a "generally licensed device" (GLD). A general license may also be issued to install and service generally licensed devices.

Regulations in 10 CFR 31.5 also require certain general licensees who possess devices containing at least 370 MBq of cesium-137, 3.7 MBq of strontium-90, 37 MBq of cobalt-60, or 37 MBq of americium-241, or any other transuranic (i.e., element with atomic number greater than uranium) to register with the NRC.

There are also general licenses for distribution. General distribution licenses are issued by NRC or by Agreement States, only authorize the commercial distribution

of byproduct materials[3] to general licensees, and do not authorize the possession, manufacturing or use of such radioactive material. Examples include sealed sources intended for devices designated for detecting, measuring, gauging, or controlling density, thickness, radiation leakage, or chemical composition, or for producing light or an ionized atmosphere; $^{3}H$ or $^{147}Pm$ sources intended for luminous aircraft safety devices; $^{241}Am$ sources for calibration and reference measurements; and certain byproduct materials in prepackaged units for use in certain in vitro clinical or laboratory tests.

General licenses are also issued to physicians, clinical laboratories, hospitals, and veterinarians in the practice of veterinary medicine, authorizing the possession and use of the following byproduct material in prepackaged units (e.g., kits, sources, and standards) for in vitro clinical or laboratory tests not involving internal or external administration of the radioactive material or radiation from it to human beings or animals (10 CFR 31.11) [14]:

- $^{14}C$, in units not exceeding 370 kBq each.
- $^{57}Co$, in units not exceeding 370 kBq each.
- $^{3}H$, in units not exceeding 1850 kBq each.
- $^{125}I$, in units not exceeding 370 kBq each.
- Mock $^{125}I$ reference or calibration sources, in units not exceeding 1.85 kBq of $^{129}I$ and 0.185 kBq of $^{241}Am$ each.
- $^{131}I$, in units not exceeding 370 kBq each.
- $^{59}Fe$, in units not exceeding 740 kBq each.
- $^{75}Se$, in units not exceeding 370 kBq each.

Within the Registration, Inspection, and Certification Program, *X*-ray machines —radiology, dental, fluoroscopy, CT, mammography, bone density, electron microscopes, etc.—and electron accelerators are registered with the state before use, after acquisition, and periodically. Registration of radiation machines servicing and services is also required.

NRC issues specific licenses to the following: (1) construction, operation, and decommissioning of commercial reactors and fuel cycle facilities; (2) possession, use, processing, exporting, importing, and certain aspects of transporting

[3]A byproduct material is a radioactive material (except special nuclear material) yielded in or made radioactive by exposure to the radiation incident to the process of producing or using special nuclear material: (1) tailings or wastes produced by the extraction or concentration of uranium or thorium from any ore processed primarily for its source material content; (2) any discrete source of $^{226}Ra$ that is produced, extracted, or converted after extraction for use for a commercial, medical, or research activity; or any material that has been made radioactive by use of a particle accelerator and is produced, extracted, or converted after extraction for use for a commercial, medical, or research activity; (3) any discrete source of naturally occurring radioactive material, other than source material, that the Commission determines would pose a threat similar to the threat posed by a discrete source of $^{226}Ra$ to the public health and safety for use in a commercial, medical, or research activity.

radioactive materials and waste; and (3) siting, design, construction, operation, and closure of waste disposal sites [15]. The licensing process includes approving the initial license, subsequent license modifications, and license renewals.

Besides applying the Registration, Inspection, and Certification Program for all kind of radiation producing machines and systems—diagnostic *X*-ray systems and their major components, radiographic equipment, fluoroscopic equipment, computed tomography (CT) equipment, cabinet *X*-ray systems, and mammography systems, etc.—Agreement States issue specific licenses for the following:

(1) Uranium and all other specific source materials[4] excluding depleted uranium used as shielding and counterweights;
(2) Radioactive materials used in sealed sources or in calibration and reference sources;
(3) Processing or manufacturing of radioactive materials for commercial distribution or industrial uses;
(4) Processing or manufacturing, and distribution of radiopharmaceuticals, including radiopharmacies;
(5) Industrial radiography;
(6) Irradiation of materials (foods, chemicals, blood, etc.);
(7) Medical, veterinary, industrial, academic, and research uses of radioactive materials (fixed gauging devices, well logging, tracer studies, nuclear laundry, portable gauging devices, in vitro and clinical laboratories, gas chromatography devices, teletherapy, gamma stereotactic radiosurgery, fixed and mobile high dose rate remote afterloading devices, nuclear medicine services, medical institutions, including hospitals, private practice physicians, nuclear powered pacemakers, etc.;
(8) Commercial waste disposal or treatment facilities, including burial or incineration, compaction, repackaging storage or transfer, and commercial treatment of radioactive materials for release to unrestricted areas; and,
(9) Exempt consumer product uses.

Any authorization, that is, registration (either as notification, registration or general license) or license (specific license) is granted on the basis of a safety assessment prepared and submitted to the regulatory authority by the person, enterprise, or organization applying for authorization, and accompanied by specific requirements and conditions to be complied.

[4]Source material means either the element thorium or the element uranium, provided that the uranium has not been enriched in the isotope uranium-235. Source material also includes any combination of thorium and uranium, in any physical or chemical form, or ores that contain by weight one-twentieth of one percent (0.05 %) or more of uranium, thorium, or any combination thereof. Depleted uranium (leftover from uranium enrichment) is considered source material.

### *10.2.4 Authorization Process*

During the authorization process, all conditions and limitations that apply, existing regulatory requirements for the practice or source, and the requirements for the safety assessment are taken into consideration.

The process of authorization is typically conducted in accordance with the workflow illustrated in Fig. 10.1. It is always up to the regulatory authority using different procedures for registration and license.

The first step for authorization is notification and, as appropriate, application for registration, or license. Some general licenses (e.g., of static eliminators, measuring, gauging, and control devices, etc.) may be effective without the filing of an application or the issuance of licensing documents; although, the particular general licensee should apply for a certificate before the receipt of the radioactive material. This applies only to a radioactive material manufactured, or initially transferred,

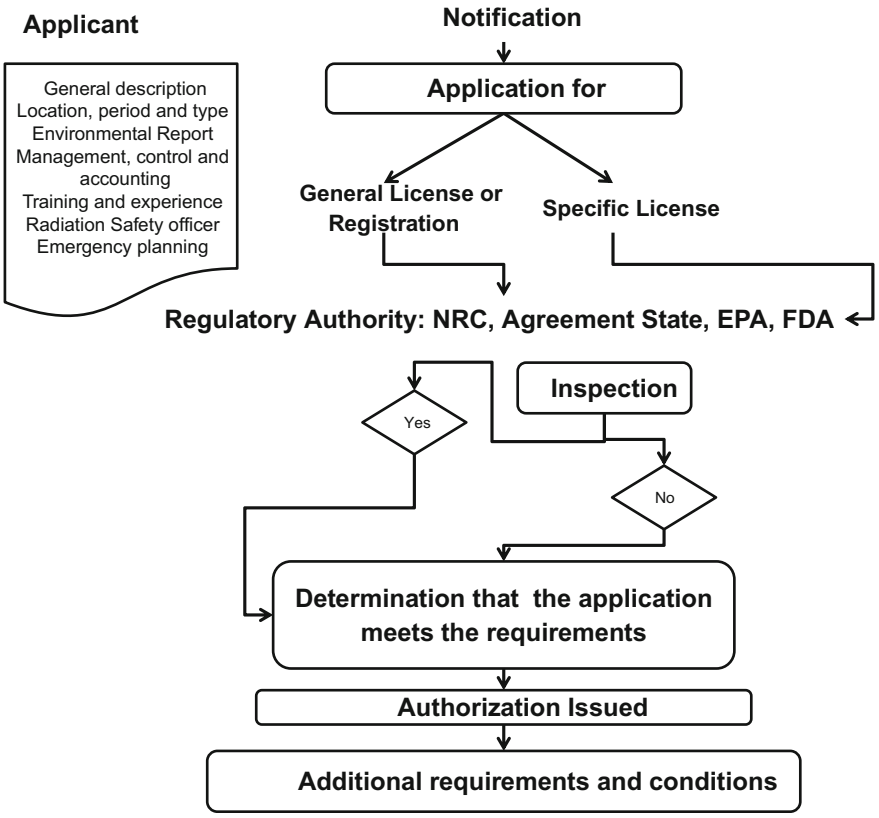


**Fig. 10.1** Process of authorization workflow

and labeled in accordance with the specifications contained in a specific license issued before.

The applicant submits a detailed demonstration of safety attached to the application, along with the measures taken for the safety and security of sources, and to protect workers and public, which are to be reviewed by the regulatory body in accordance with settled procedures. The extent of the control required is commensurate with the potential magnitude and nature of the existing hazards. Thus, for example, if a given dental *X*-ray machine is manufactured according to the requirements, a registration may just be required; whereas a commercial waste disposal facility will possibly require an authorization for commencement of construction with any appropriate conditions to protect environmental values, and other for operation. A specific license is needed for waste collection as well.

Usual information for the assessment includes expected and potential exposures to workers and members of the public, as appropriate; general and thorough descriptions of the sources, equipment and facilities; administrative controls and provisions relating to organization and management, procedures, record keeping, material control and accounting, etc., necessary to assure safe operations; as well as the appointment of a radiation safety officer who is qualified by training and experience, and who is available for advice and assistance on radiation safety matters. It also comprises completion of safety evaluations of the proposed uses of radioactive material, the name, address, and title of the person responsible for the safety and security of sources, and the name and qualification of all persons responsible for operation, including the corresponding training and experience, if appropriate [16].

Depending on the type of authorization, a floor plan of the premises showing relevant information may be required, such as source locations, beam directions, shielding materials, adjacent rooms and areas, occupancy factors, storage areas, etc., and a comprehensive report indicating that all applied requirements for operation, maintenance, and source disposal have been met, including calculations supporting the material and thickness of shielding, and the estimated doses.

When authorization is related to a specific stage, for example, an early site permit or application to construct or operate a nuclear reactor, results of geotechnical investigation programs, or acceptance tests, might be required. The Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants includes location specifications, site characteristics and parameters that could affect the safe design and siting of the plant (flooding, tsunamis, storms, tornados, etc.), local meteorology, and basic geologic and seismic information. The applicant's legal authority to determine all activities within the designated exclusion area, and control and the population distribution, may be required, among others [17].

One or more inspections might be necessary during the review to gather complete information about the safety requirements accomplishment, protection and safety program, and to confirm the adequacy of such programs.

The regulatory authority can also take formal actions through the review, such as sending back all or part of the documents, call for meetings to address outstanding issues, carry out additional consultations or inspections, etc., which might result in

either granting the authorization or its refusal. Failure to take recommended actions within a reasonable time is ground enough for a negative response [7].

When a specific phase, equipment, facility, activity, or use is authorized, the authorization also establishes the additional requirements, conditions or limitations to be applied. Authorizations are effective for the limited period of time stated on the authorization. Any subsequent authorization, amendment, renewal, suspension, or revocation is then undertaken as stated by established procedures. Procedures include the requirements for the timely submission of the corresponding applications.

Authorized facilities, equipment, activities, or specific operations are then subject to ordinary inspections by the regulatory authority. Ordinary inspections are also required to maintain control on operations regarding the sources or facilities, for example, the source disposal to a convenient facility, the equipment commissioning or the facility decommissioning.

## 10.3 Technical and Verification Requirements for Safety

Technical requirements should commensurate with the nature and magnitude of the sources, and the complexity of the operations involved. The management system considers generic safety requirements for the sources security, and specific defense in depth and good engineering practices requirements, proportionate to the level and likelihood of planned exposures. Compliance verification is a critical component in the control of all types of exposure.

### *10.3.1 Requirements for the Security of Sources*

As learned from past accidents—abandoned or inappropriate stored sources found and taken home, stolen sources, and sources that fell during transport—the lack of control over the sources might lead to fatalities, severe injuries, and economic losses. Hence their security has been and continues to be, one of the most important prerequisites for their safety.

Safety measures and security measures have a lot in common, and should be designed and implemented in an integrated manner, so that security measures do not compromise safety and safety measures do not compromise security. Such measures include access control and/or surveillance, authorization, and control of transfer and transportation, and related information protection.

The security requirements for the sources are meant for preventing theft, damage, or any other unauthorized use of sources, i.e., for keeping them secure and under control. They consist of appropriate source storage and safeguarding conditions to deter, detect, delay, and respond to any unauthorized access, or the theft, loss, or unauthorized use or removal of radioactive sources during all stages of their

management [18]. These requirements are applicable to the entire source life cycle, including manufacturing, supply, receipt, storage, use, transfer, import, export, transport, maintenance, and disposal.

Rooms, cabinets, and areas where radiation sources are used or stored, are to be provided with warning labels and physical barriers like doors with keys, fences, walls, other buildings, etc. Fences can be only for demarcation or can be combined with intrusion detection systems. It is recommended that door locks and hinges offer some resistance to forcible attack. Measures to detect and prevent an intrusion include audible and visible warning signals, alarms, security seals, and effective illumination, and so on. Safety interlocks and/or key actuated controls are recommended since the source will be moved or shielded when they are tripped.

A proper design and manufacture of radiation sources, consistent with regulations, is significant to security. Authorization of possession and use should be required for the supply, transfer, receipt, import, and export of radiation sources. Containment and shielding, consistent with the activity and type of sources, are important requirements for their use and transportation. Although sources in Category 5 are less dangerous (see Table 10.1); they could give rise to doses in excess of the dose limits if not properly controlled.

Authorized users and manufacturers are responsible for the safe and secure management of radiation sources, even during their transportation and/or transfer to other organization, e.g., for disposal or reuse. Hence, they are to maintain and update records about the actual location and movements of sources, as well as accurate inventories of all radioactive materials and radiation producing equipment.

Inventories includes the make, model, number, and serial number of sealed sources and/or radiation producing machines, as well as the radionuclide and activity of each unsealed source, along with its location and date of use. Container technical descriptions and forms of source receipt and transfer should be kept as long as the sources are in inventory.

During transportation, radiation sources must be in containers, metal cages, or transportation packages; containers, cages, and packages should be labeled, locked, and secured to the vehicle to prevent accidental loss, tampering, or unauthorized removal of the sources.

Transfers and disposals are to be previously approved and properly documented. A radiation source never should be abandoned unless it is considered irretrievable, in which case the site is to be properly identified and secured following approved procedures, e.g., immobilization and sealing [19]. These measures are important when the sources are used in field applications (e.g., well logging).

### *10.3.2 Defense in Depth*

The concept of defense in depth is a well-established principle for the design of equipment and facilities where radiation sources are used. The aim is that a single equipment fault or human mistake should not directly result in an accident. Defense

in depth is defined as "a hierarchical deployment of different levels of diverse equipment and procedures to prevent the escalation of anticipated operational occurrences, and to maintain the effectiveness of physical barriers placed between a radiation source or radioactive material and workers, members of the public or the environment, in operational states and, for some barriers, in accident conditions" [6] ["Reproduced with permission by the IAEA"].

In other words, defense in depth is a multilevel system of sequential, independent provisions for protection and safety, commensurate with the likelihood and magnitude of potential exposures, such that if one level of protection were to fail, the subsequent independent level of protection would be available, for the purposes of [1] ["Reproduced with permission by the IAEA"]:

(a) "Preventing accidents;
(b) Mitigating the consequences of any accident that do occur; and
(c) Restoring sources to safe conditions after any such accident."

Defense in depth includes the use of access controls, physical barriers, redundant, and diverse key safety functions, and emergency response measures. Examples are: design fail-safe features in *X*-ray generators which cause the beam port shutters to close upon the failure of a key component; one or more safety interlocks and barriers to prevent the accidental entry in the irradiation chamber in the design of irradiator facilities; multiple and redundant access control devices to preclude access to an area of radiation hazard; technological and structural barriers as shielding, surrounding walls or cabinets, labyrinths, air locks, locker rooms, decontamination areas, air filters, and so on, to reduce and restrict the levels of radiation and/or contamination; confinement technology such as fume cupboards, glove box, hot cells, etc., with forced ventilation and remote handling devices to work with radioactive materials; reactor vessel, fuel cladding, coolant systems, pressure boundary components, and containment structures to prevent radioactive release in nuclear reactors; and many more commensurate with the type, magnitude, and likelihood of exposure.

Containment barriers—absorbent materials, sealed cans, shielded containers, seals, and fastening devices, etc.—used for packaging during transportation of radioactive materials are also examples of defense in depth to prevent releases in case of incidents or failure.

Defense in depth also includes measures to ensure that tasks are assigned only to fully qualified personnel, that procedures are followed, emergency procedures are available and rehearsed regularly, and that all equipment is correctly used, maintained, and tested regularly.

### 10.3.3 Good Engineering Practices

Registrants and licensees, in cooperation with other responsible parties, should ensure, as applicable, that the siting, location, design, construction, assembly, commissioning, operation, maintenance, and decommissioning of facilities, or parts thereof, are based on sound engineering practices that provide appropriate, cost-effective, and well-documented solutions to meet safety requirements and compliance with applicable regulations. Good engineering practices put emphasis on compensate human errors and reduce the probability of failures that may conduct to accidental exposures through design. As part of the management system, all engineering decisions that have a major impact on radiation safety are also reviewed by radiation protection specialists.

More reliable technologies designed to reduce the maintenance on radioactive components, to reduce radiation fields, to diminish repair and removal times, and to accommodate remote and semi-remote operation, maintenance, and inspection, are means to minimize the dose and the time spent in radiation areas.

Examples are self-shielded computerized tomographic (CT) scanners for medical imaging; screening for lung cancer with low-dose spiral or helical CT; compact self-shielded gamma irradiators for agriculture and medicine, and new high-power accelerators with direct electron *X*-ray conversion for industrial irradiation.

### 10.3.4 Monitoring and Verification of Compliance

Security, defense in depth, and sound engineering requirements, are important technical standards which fulfillment is necessary, but not sufficient, to achieve the highest level of security and protection. The assumptions made when planning and designing, in addition to the probabilistic nature of failures, require systematic monitoring and verification to maintain the dose as low as reasonably achievable.

Regular measurement of specific parameters at the source and the environment (source and environmental programs), verification, testing, and calibration by certified third parties of measuring instruments—routine and emergency—as well as recording for compliance and reporting, are then required. Data from occupational exposure and public exposure, workplace and environmental surveillance, personnel training, maintenance of equipment and systems, calibration of instruments, etc., are recorded and documented. Area monitoring and calibration records are usually kept for five years. Worker exposure records are kept for the working life and, afterwards, at least until 75 years of age, but no less than 30 years, after the termination of work.

"Some facilities and activities—hospitals or research institutes using short lived radionuclides—may not require an environmental monitoring program; some—small nuclear installations or nuclear medicine departments using radionuclides for diagnostic purposes—may require routine monitoring at the source, but only

occasional checks on environmental levels; and others—most nuclear installations, large nuclear medicine departments—require continuous and comprehensive monitoring of both source and environment" [20] ["Reproduced with permission by the IAEA"].

The sources and equipment used for medical exposure are also subject to calibration in a regular basis following internationally or nationally accepted protocols. Doses administered to patients in the course of diagnostic and treatment are controlled and compared, as appropriate, to existing local or national reference levels.

The regulatory authority verifies through surveys and/or inspections that the source and environment monitoring programs are in compliance with the authorized limits on discharges. Also, that the assumptions, models and parameters that were used in the licensing process are consistent with the actual specific conditions. Instruments used by the regulatory authority to verify compliance with regulatory requirements must be appropriate for use and calibrated at required frequencies.

### *10.3.5 Safety Assessment*

Safety is to be assessed for all facilities and activities, consistent with a graded approach [21]. For nuclear reactors, radioactive waste, and fuel reprocessing facilities, as well as for certain irradiators, usually various safety assessments are performed by stages—e.g., early site permit, design, commencement of construction, operation, decommissioning, etc.—some facilities—hospitals or research institutes, large nuclear medicine departments—may require a safety assessment for design and operation, and a final safety assessment for closure; and others need safety assessments only for manufacture, distribution, or operation.

Safety assessments cover the means that the normal and potential exposures could occur considering all probable events—external, and directly related to the source and their associated equipment—the probability and magnitude of expected exposures in different foreseeable and reasonable situations using deterministic and also probabilistic methods; the safety measures necessary to control the hazard; and the evaluation of the design and engineered safety structures, systems, and components needed to mitigate or prevent these events, to demonstrate that they fulfill the safety functions required of them.

The primary purpose of the safety assessment is to determine whether an adequate level of safety has been achieved for the given facility or activity. Also, to decide if the basic safety objectives and safety criteria established by the designer, the operating organization and the regulatory authority have been fulfilled [21].

Elements to be assessed are summarized in Fig. 10.2. The assessment of all or some elements, as appropriate, will determine whether the provided defense in depth through physical barriers, systems to protect the barriers, and administrative procedures is adequate for the facility or activity. Long-term safety is of particular concern when aging effects might develop and affect safety margins, for example,

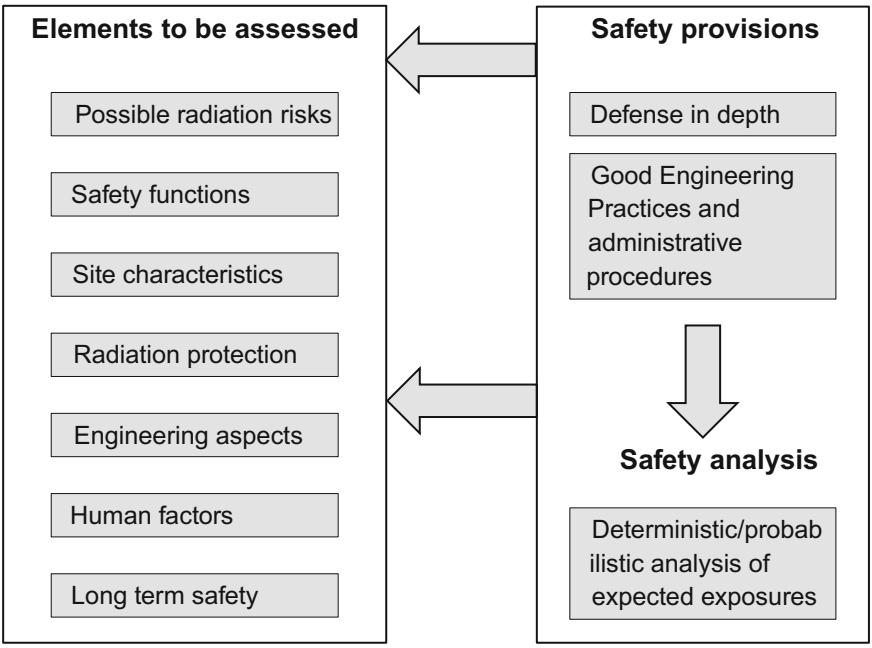


**Fig. 10.2** Safety assessment

in a radioactive waste disposal facility. Computer codes used for the safety analysis need to be verified, tested, and validated.

Reassessment of facilities and activities that continue over long periods of time, on a regular basis, are also necessary to assure that the technical specifications and conditions are consistent with any possible changes in circumstances, to identify new opportunities for improving protection and safety—the application of new standards or new scientific and technological developments—changes in site characteristics, and modifications to the design or operation, and also the effects of aging.

## References

1. International Atomic Energy (2014) Radiation Protection and Safety of Radiation Sources: International Basic Safety Standards, IAEA Safety Standards Series GSR Part 3. IAEA, Vienna
2. UNSCEAR (2012) Annex A—Attributing health effects to ionizing radiation exposure and inferring risks. UNITED NATIONS, New York 2015
3. ICRP (2007) The 2007 recommendations of the international commission on radiological protection, Ann ICRP, vol 37, no 2–4, April–June 2007
4. International Atomic Energy (2006) Fundamental safety principles, Safety Standards Series No. SF-1. IAEA, Vienna

5. International Atomic Energy (2006) The management system for facilities and activities, Safety Requirements No. GS-R-3. IAEA, Vienna
6. International Atomic Energy (2007) IAEA safety glossary, terminology used in nuclear safety and radiation protection—2007 Edition. IAEA, Vienna
7. International Atomic Energy (2007) Notification and authorization for the use of radiation sources ((Supplement to IAEA Safety Standards Series No. GS-G-1.5), TECDOC-1525. IAEA, Vienna
8. NRC Regulations Title 10 (2001) Code of Federal Regulations, Part 40, Exemptions for uranium contained in aircraft counterweights, July 20, 2001
9. Florida Administrative Code (2007) Chapter 64E-5, Rule number 64E-5.511, Registration of radiation machines DH 03/2007
10. International Atomic Energy (2005) Categorization of radioactive sources. Safety Guide No. RS-G-1.9. IAEA, Vienna
11. International Atomic Energy (2006) Dangerous quantities of radioactive material (D-values), Emergency Preparedness and Response, EPR-D Values 2006. IAEA, Vienna
12. NRC Regulations Title 10, Code of Federal Regulations (2007, 2008) Part 35, Medical use of byproduct material, Oct. 1, 2007 and Jul. 23, 2008
13. NRC Regulations Title 10, Code of Federal Regulations (2013) Part 31, General domestic licenses for byproduct material, July 25, 2013
14. NCR Regulations Title 10, Code of Federal Regulations (2014) Part 31, Section 31.11 General license for use of byproduct material for certain in vitro clinical or laboratory testing, December 19 2014
15. U.S.NRC (2015) "Licensing," United States Nuclear Regulatory Commission, 2015. http://www.nrc.gov/about-nrc/regulatory/licensing.html
16. Suggested State Regulations for Control of Radiation (2015) Conference of radiation control program directors, 2015. http://www.crcpd.org/SSRCRs/default.aspx
17. U.S.NRC (2014) Standard review plan for the review of safety analysis reports for nuclear power plants: LWR Edition (NUREG-0800, Formerly issued as NUREG-75/087), United States Nuclear Regulatory Commission, 2014. http://www.nrc.gov/reading-rm/doc-collections/nuregs/staff/sr0800/
18. International Atomic Energy (2009) Security of Radioactive Sources Implementing Guide, Nuclear Security Series No. 11. IAEA, Vienne
19. U.S.NRC (2015) PART 39—Licenses and radiation safety requirements for well logging, United States Nuclear Regulatory Commission, 2015. http://www.nrc.gov/reading-rm/doc-collections/cfr/part039/full-text.html
20. International Atomic Energy (2005) Environmental and source monitoring for purposes of radiation protection, Safety Guide No. RS-G-1.8. IAEA, Vienna
21. International Atomic Energy (2009) Safety assessment for facilities and activities, General Safety Requirements Part 4. IAEA, Vienna

# Chapter 11
# General Principles of Radiation Protection

For pragmatic reasons and mostly with the intent of regulation, the ICRP adopted in the 1950s a linear no-threshold (LNT) dose–response relationship—a model indicating that there will be some risk even at low doses—though this hypothesis is still not deemed proven. In the 1990s, the LNT model was modified by a dose-rate effectiveness factor (DDREF), to account for an apparent decrease in the effectiveness of low-LET radiation in causing a biological end-point (e.g., cancer) at low doses and dose rates, compared with observations made at high, acutely delivered doses [1].

Now, several studies suggest a lack of adherence to an LNT response, although the interpretation of the available evidence is controversial. Nonetheless, based on current scientific knowledge, the health effects associated with radiation exposure, i.e., stochastic effects, such as cancers, cannot be unequivocally attributed to radiation exposure [2, 3].

The various plausible dose response relationships between cancer risk and exposure, in the ranges of low and very low doses, are illustrated in Fig. 11.1. They are known as (a) supralinear; (b) linear non-threshold (LNT); (c) linear–quadratic; (d) threshold; and (e) hormetic [3] ["From Sources and Effects of Ionizing Radiation, 2012 Report to the General Assembly with Scientific Annexes, Annex A —Attributing health effects to ionizing radiation exposure and inferring risks, by UNSCEAR, ©2015 United Nations. Reprinted with the permission of the United Nations."]

At moderate and high doses, an increased frequency of occurrence of certain health effects can be confidently attributed to radiation exposure. There is sufficient evidence, knowledge, and scientific consensus regarding causal relationships, to be able to predict relatively accurately tissue reactions and their possible severity from exposures at high doses, and an increased risk of stochastic effects from exposures at moderate doses.



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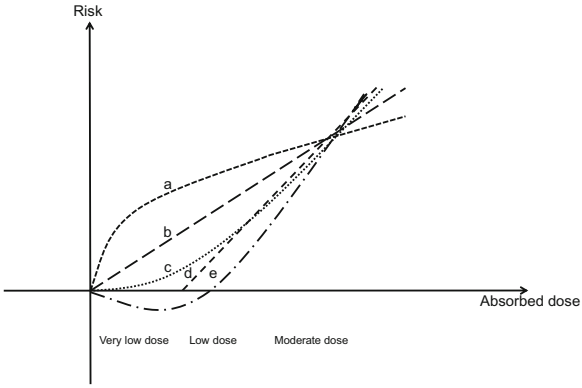


**Fig. 11.1** Various possible dose–response relationships

In contrast to moderate and high doses, the confidence substantially decreases at lower doses. At lower doses—100 mGy or less—projections of the absolute number of cancer cases in a population have less and less information value and can be increasingly misleading. Thus, the following hypotheses cannot be convincingly verified or falsified [3] ["From Sources and Effects of Ionizing Radiation, 2012 Report to the General Assembly with Scientific Annexes, Annex A—Attributing health effects to ionizing radiation exposure and inferring risks, by UNSCEAR, ©2015 United Nations. Reprinted with the permission of the United Nations."]:

- The currently observed response in the moderate-dose range can be extrapolated linearly down to zero incremental dose above that from normal natural background radiation (this would be a linear non-threshold [LNT] relationship);
- The risk at low and very low doses is substantially higher than expected from a LNT relationship due to, e.g., the lack of activation of repair mechanisms or bystander effects (this would be a supralinear relationship);
- The risk at low and very low doses is substantially lower than expected from a LNT relationship; a threshold dose below which there is no risk of harm, or a beneficial effect—an increase of immune defense—from exposure to low doses or very low doses, are expected (this would be a threshold or hormetic relationship).

Based on current scientific knowledge, health effects attributable to radiation exposure are not distinguishable from the effects that arise from other causes. Thanks to developments in the radiobiology and molecular biology fields in the past decade, a number of laboratory data from cancer studies suggest that cancer risk for low and very low doses may be less than estimated by the LNT model, even after employing a DDREF, and that a threshold dose is not improbable [4, 5]. These studies also propose that cell defenses make possible to reduce or prevent the harmful effects of ionizing radiation [6, 7]. A better understanding of biological mechanisms is still looked-for, and some of these mechanisms—adaptive response, apoptosis, genetic predisposition, bystander effects, genomic instability, etc.—deserve particular attention.

For example, health studies of populations in places such as Ramsar, Iran where the levels of natural radiation in one particular district can reach up to 260 $mSv \cdot y^{-}1$, have failed to reveal any ill-effects attributable to radiation. Moreover, preliminary results from these studies suggest that exposure to high levels of natural background radiation can induce an adaptive response in human cells [8, 9]. Conversely, some epidemiological studies of workers exposed to low doses of radiation prompt to slightly elevated cancer mortality rates, specifically of leukemia [10, 11].

The debate over the effects of low level radiation is still contentious and unsettled [12], but inferred risks remain and protection measures against such risks are required [13, 14]. So, the main objective of radiation protection has been and continues to be, protect individuals, society as a whole and the environment against potential harmful effects of ionizing radiation [15].

To achieve its objective, the sole application of permissible limits is until present considered not enough. As stated by the latest ICRP recommendations [1], there are three general radiation protection principles: justification, optimization, and application of dose limits. Justification and optimization are applied to all exposure situations. Dose limits apply to planned exposure, both occupational and public, but not to medical exposures of patients.

## 11.1 Justification

This principle establishes that, "by introducing a new radiation source, by reducing existing exposure, or by reducing the risk of potential exposure, one should achieve an individual or societal benefit that is higher than the detriment it causes" [1] ["Reprinted from ICRP Publication 103 with permission of the ICRP."]. The consequences include other potential risks, and the costs and benefits of the activity.

For planned exposure situations, justification means that any planned exposure should produce sufficient net benefit to the exposed individuals or to society to offset the radiation detriment it causes. Justification is an important element to consider during the authorization of facilities and activities; for example, to authorize a waste disposal facility, it is important to consider all its societal, economical, and safety impacts. The principle of balancing benefit and detriment is not unique to radiation safety, but while often the balancing is generally done implicitly, in our case, before the regulatory authority can authorize a facility or an activity, it requires an explicit demonstration of a positive net benefit.

When the exposure can only be controlled by actions to modify the pathways of exposure and not by acting directly on the source, as in emergency and existing exposure situations, any decision to reduce doses, which always have some disadvantages, should do more good than harm [15]. For example, a measure as disruptive as an evacuation should be justified by the dose averted. Otherwise, sheltering is preferred. The Government of Japan recommended the evacuation of about 88,000 people around the Fukushima power plant, and the sheltering in their

own homes of about 62,000 other people. Evacuation averted effective doses to adults of up to about 50 mSv and absorbed doses to the thyroid of 1-year-old infants of up to about 750 mGy [16] ["From Sources and Effects of Ionizing Radiation, 2013 Report to the General Assembly, Volume I Scientific Annex A. Levels and effects of radiation exposure due to the nuclear accident after the 2011 Great East Japan earthquake and tsunami, by UNSCEAR, ©2014 United Nations. Reprinted with the permission of the United Nations."].

There are certain exposures that are unjustified without further analysis, unless there are exceptional circumstances. For example, except for justified practices involving medical exposure, activities that result in an increase in activity, by the deliberate addition of radioactive substances or by activation, in food, feed, beverages, cosmetics, or any other commodity or product intended for ingestion, inhalation or percutaneous intake by, or application to, a person are unjustified. Also unjustified is the frivolous use of radiation or radioactive substances in commodities and products, such as toys and personal jewelry or adornments [1, 15]. Human imaging using radiation for occupational, legal, or health insurance purposes undertaken without reference to clinical indications, or for theft detection purposes, or for the detection of concealed objects for security or antismuggling purposes, although has been usually deemed not justified, can be justified by the government for reasons including national security [17].

Medical exposures are intentional but, as a prevailing criterion, they do more good than harm to the patient. It can occur in diagnostics and in image-guided interventional and/or therapeutic procedures. The responsibility for the justification of the use of a particular procedure at the individual level falls on the relevant medical practitioners, who take into account the particular objectives of the exposure recommended by referring physicians, the clinical circumstances and the characteristics of the patient, as well as the benefits and risks of available alternative techniques. When the patient is pregnant, breast-feeding, or pediatric, it is also a requirement to consider the appropriateness of the request and the urgency of the procedure [15].

But there is also a need for generic justification of a given radiological procedure by the health authority in conjunction with appropriate professional bodies. This applies to the justification of new technologies and techniques, for example, the FDA, or an FDA-approved state certifying agency, certifies mammogram facilities in the United States under a law called the Mammography Quality Standards Act (MQSA). In 2011, after determining that there was a reasonable assurance that new 3-D mammography devices were safe and effective for their intended use, the FDA approved the first device that provides three-dimensional (3-D) images of the breast for breast cancer screening and diagnosis.

Medical radiological screening of asymptomatic population groups is considered unjustified, unless it is part of an approved health screening program for the early detection of disease [1]. Screening tests differ from diagnostic studies in that they are usually applied in the evaluation of healthy individuals. Radiological screenings should be specifically approved by the health authority in conjunction with professional bodies. For example, in the United States, the U.S. Preventive Services

Task Force (USPSTF) now recommends against routine screening of women aged 40–49 years, and recommends screening mammography every two years, as opposed to the annual screening advocated by many breast cancer specialists, for all women aged 50–74 years [18]. The consideration is based on the high risk of over diagnosis, followed by harmful treatment—surgery, radiation therapy, and or chemotherapy—of a cancer that would not have become a threat to a woman's health.

## 11.2 Optimization

The principle of optimization establishes that the level of protection should be the best under the prevailing circumstances, maximizing the margin of benefit over harm. Optimization is source related and applies to the number of people exposed, to the magnitude of individual doses and the likelihood of potential exposures in all exposure situations.

Constraints used in planned exposure situations, and reference levels in emergency and existing exposure situations, as well as in medical exposures, provide the desired bound for the optimization process. They are also used as a benchmark to assess the suitability of the optimized protection strategies implemented.

Optimization can be applied to the design, operation, and decommissioning of sources, facilities, or activities, to the disposal of radioactive waste, and to emergency or remedial actions. Optimization is often accomplished by actions that improve the working conditions as well; thus its goals can be achieved together with an increased efficiency at a minimum or no practical net financial cost.

As shown in Fig. 11.2, optimization is an iterative and constant process of analyzing whether or not reducing the dose is practicable, and if it is, reasonably restrict the magnitude and probability of exposures before they occur. In other words, it is a systematic process of questioning if the best was done concerning security and safety in the prevailing circumstances, considering all relevant factors [19]:

- The nature, magnitude, and likelihood of exposures;
- The total detriment from actual and potential exposures;
- The cost of protection; and
- Other reasonable harms.

The evaluation will always depend on the nature and magnitude of the sources. For example, to optimize the exposure from an *X*-ray machine, the first concern may be on the specifics of the device—i.e., if it is compliant with the applicable standards of the International Electrotechnical Commission (IEC), and/or the International Organization for Standardization (ISO), and/or the standards accepted by the regulatory authority. Once the suitability is established, to optimize its safety, attention should be paid to:

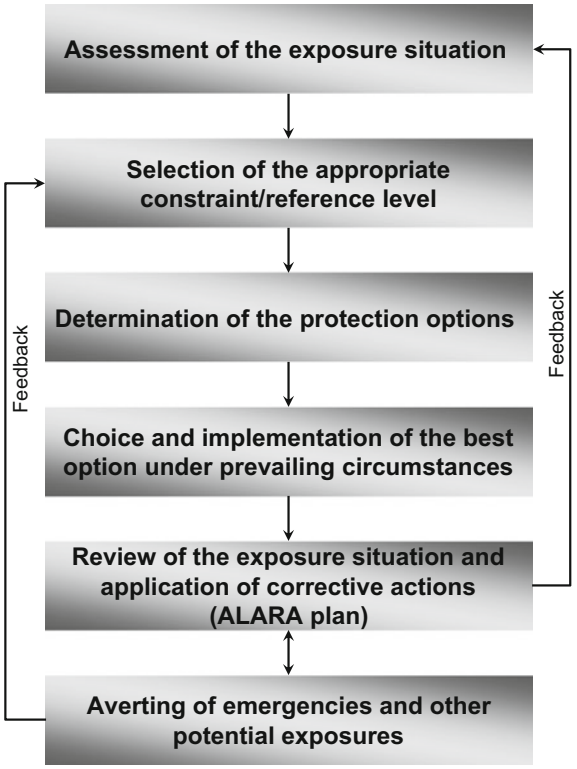


**Fig. 11.2** Common stages of the optimization process

- The specifics of the room and location where the apparatus is operated;
- The operating procedures and protocols; and
- The qualification and training of certified operators.

In a large irradiation facility, apart from reviewing the accomplishment of specific design requirements and operation procedures, the assessment may also take into account issues like, for example:

- Performance of the hydraulic, ventilation, and transport systems;
- Redundancy of interlocks and security devices to prevent non-authorized accesses and/or return the source to its safety position;
- Effectiveness of the confinement and mechanical protection of sources to prevent its interference and damage;
- Automation and supporting devices like built-in area monitoring meters and alarms;
- Appropriateness of the equipment and instrument maintenance and servicing;
- Replacement, storage, and disposal of the sources.

For a mobile radiographic device used in field conditions, the essential aspects to prevent unnecessary exposures of workers and the public might be the availability

and adequacy of compliant operating procedures, and the physical control and safety storage of the sources.

At a nuclear power plant, optimization is still a more complex process. The assessment in this case should address and evaluate issues like specific problems related to the aging of existing plants, safety systems maintenance and repair (in-service inspections), machinery breakdown prevention, spent fuel safety (storage, transport, etc.), environmental effects of nuclear power generation, radioactive waste management and treatment, decommissioning, projected doses in the future, etc. The NCRP Report No. 120 [20] presents the most recent quantitative methods used in the decision-making process to organize, direct, and administer a program aimed at keeping doses as low as reasonably achievable (ALARA) at nuclear power plants.

Whatever the complexity, there are two main levels of assessment [19]. The first consists of a global evaluation of the exposure to identify the major areas for improvement and to check the overall effectiveness of an optimization program, if one is already in place. The second involves a detailed analysis of specific jobs to examine the factors that contribute to the associated doses or risks, and to determine the appropriate actions that could be taken for its reduction.

The global evaluation entails in analyzing the factors characterizing the specific exposure situation, including:

- The level of collective dose (predicted or actually received);
- The monetary value of collective dose, if applicable;
- The distribution of individual doses;
- The cost of protection;
- Probable impacts to members of the public or to a particular crucial area that could not be changed;
- Environmental consequences, if any;
- Possible doses to future generations.

Levels of collective doses and distributions of individual doses, especially maximum individual doses and the number of persons exposed, can be obtained from a variety of available data, for example: individual doses monitored during a period of time or while carrying out a specific job or task; past events records; models of good practice for the given job in other facilities; etc.

A detailed analysis of specific jobs for optimization purposes includes, at least:

(a) All performed operations, including type, frequency, sequence, and duration of tasks; technological and safety equipment required, level of automation, number of individuals involved, etc.;
(b) Detailed procedures, e.g., distance from the source, length of exposure, body parts most likely exposed, personal protective equipment worn, etc.;
(c) Measured and/or realistic expected dose rates, levels of surface contamination, aerosol activity, and individual doses;
(d) Possible deviations according to individuals and groups, and reasonably foreseeable potential exposures with basic explanation;

(e) Resources for protection and safety available and in use;
(f) Personnel qualification and training;
(g) Level of compliance with requirements;
(h) History of failures and incidents, causes and consequences;
(i) Rate of recurrence of corrective actions and results;
(j) Effect of radiological protection actions on other risk factors.

It is important to describe, as precisely as possible, the relevant radiation sources, the estimated doses and probabilities of potential exposures in different event sequences, and the specific constraints or reference levels established by the regulatory authority, the management or operator, as well as the radiological protection measures taken to meet such constraints or levels.

Dose constraints are applied in the process of optimization of planned exposure —occupational and public—and risk constraints, in the optimization of potential exposure. Constraints should always be chosen below the applicable limit—occupational exposure, public exposure—to restrict the range of options to be considered during the optimization, but never to establish a second boundary for safety [1].

For example, industrial radiographers are the most likely group of workers to receive doses approaching relevant dose limits (20 mSv); thus it is important to establish dose and dose rate constraints, as well as investigational levels for unusual exposures, to guarantee that radiation exposure, both workers and the public, is kept below the applicable limit during normal operation, maintenance, decommissioning, and in emergency situations. The recommended level for optimization of protection of industrial radiographers is 5 mSv (or less) per year [21].

Likewise, reference levels are applied in the process of optimization of emergency and existing exposure situations to identify the range of options in implementing protective or remedial actions to prevent exposures in specific circumstances. The optimization analysis is thus conducted through the planning processes considering [1]:

- The nature of the exposure and the practicability of reducing or preventing the exposure;
- The benefits from the exposure to individuals and society or the net benefit of avoiding preventive or protective actions that would be detrimental to living conditions, as well as other societal criteria related to the management of the exposure situation;
- National or regional attributes and preferences, together, where appropriate, with a consideration of international guidance and good practice elsewhere.

In medical exposures are also used dose constraints to optimize the protection of volunteers exposed for biomedical research purposes and of caregivers and comforters. Reference levels are applied as well to provide the patient with the minimum necessary dose to achieve the desired clinical objective. These reference levels are used as a constriction for a specific examination or procedure, but are not related to individual patients.

Optimization can be used to make any decision, from day-to-day operational issues to the most important modification to plant design or operation, medical procedures, procedures to prevent or mitigate the consequences of incidents and emergencies, etc. Investigation levels, such as individual doses, intakes, dose rates, or contamination levels, can also be set as a result of an optimization study. They are useful indicators to manage the overall performance of a job in comparison with the predictions, or in comparison with the best practices, and to trigger a reassessment when it is needed.

Optimization is indeed sound judgment; therefore, the methods used for the analysis can range from a simple common sense approach to complex quantitative techniques based on cost–benefit analysis. Whatever the method, the most important is to consider all factors, avoid omissions, oversights, and biases which could affect radiation protection, have alternatives, choose the best option on an informed basis, and implement it through an effective optimization plan, bringing together all requirements of time, distance, shielding, training, good practices, and well-designed and well-managed operations. The goal is to achieve a reasonable balance between the needs for dose or risk reduction, and the needs to maintain production and the costs involved [19].

The optimization outcome is a reduced dose or risk, at a minimum or no cost, and more effectiveness. The actions for these achievements vary from administrative adjustments—a better work planning or general worker education—to additional protective equipment, and major operation and design amendments to the facilities or equipment.

A wide range of aid techniques is available to help in the optimization process. Some of them are drawn from operational research, some from economics, and some from engineering. A summary of the techniques most frequently used is presented next, although other decision-making techniques may be helpful as well.

### *11.2.1 Analytical Tree*

A good start for an optimization analysis is the analytical tree. The analytical tree, also called tree analysis or tree diagram, is a technique for depicting a complete system, or subsystem, and its interrelationships. This technique is a useful tool not only for investigating unwanted events, but to evaluate the radiation protection program effectiveness and appropriateness [22, 23].

The analytical tree is used to break down a task or objective into its components and interrelations, developing each branch in detail to reach the basic elements. This technique is totally qualitative. Just to give an idea, Fig. 11.3 is an example depicting the measuring instrument part of an analytical tree [24].

The first task is to write a statement of the goal, project, plan, problem, or whatever is being studied, at the top of the tree. As seen from the example in Fig. 11.3, the goal to achieve—having appropriate measuring instruments to accomplish the radiological protection program—is at the top. Next, you write all

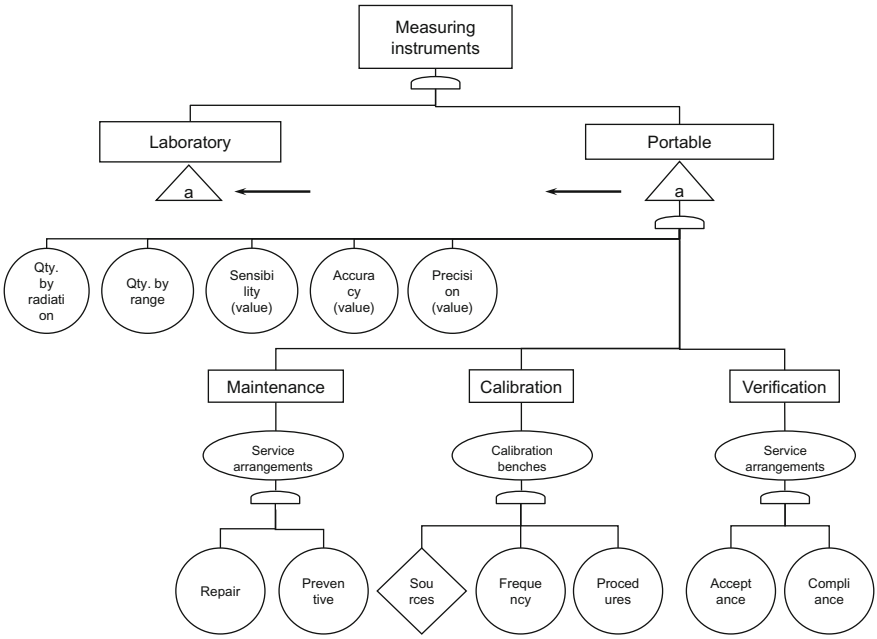


**Fig. 11.3** Example of an analytical tree as part of the measuring instruments component

the elements that must be considered from top to bottom. For instance, there are two broad categories: laboratory and field instruments that answer the question: what type of instrument is needed?

In the example, portable monitoring instruments are broken down by radiation, range, sensibility, accuracy, and precision as specific features, while laboratory instruments have their own separate tree chart. The transfer symbol, indicated by a triangle with a horizontal arrow pointing forward, shows an input from another part of the tree, and the triangle with a horizontal arrow pointing away from it shows an output to another part of the tree.

Logic symbols commonly used to denote events, components, or conditions in the analytical tree are the rectangle, circle, diamond, and ellipse. The rectangle is a general component or condition. The circle is a terminal basic or specific component, item, or constituent requiring no further development. In the example in Fig. 11.3, needed or existing instrument quantities by type and measurement range are provided, along with specific values for sensibility, accuracy, and precision.

Rectangles are also used in Fig. 11.3 to represent new resulted components to the question: what needs to be done to the instrument to make it reliable? Maintenance, verification, and calibration are then broke down further. The diamond is an undeveloped terminal component because of lack of information or resources, or to avoid redundancy. The diamond in Fig. 11.3 was used to avoid

redundancy with calibration sources; this component was to be developed in a different chart.

The ellipse is a conditional component which applies constraints on a logic gate or output. In Fig. 11.3, services arrangements are a critical constraint to instrument maintenance, if the facility does not provide such services locally. The presence and suitability of a calibration bench is also essential to calibration.

The lines connecting elements in an analytical tree are called "branches" and the elements itself are called "nodes". Each node can contain a value, a condition, or represent a separate data structure.

Optimization does not mean minimization of dose; it means the best level of protection that can be achieved under the prevailing circumstances, and analytical trees provide highly effective work breakdown structures within which to explore possible outcomes for several options.

Analytical trees can be particularly useful for the analysis and development of radiation protection programs for specific applications, or to improve the effectiveness of an existing program. To not forget or lose track of one or some components, it is recommended to brainstorm possible answers to each goal, problem, task, or condition and do a "necessary and sufficient" check. A "necessary and sufficient" check will test if all the items at a given level are necessary and sufficient for the level above [25].

### *11.2.2 Cost-Effectiveness Analysis*

Cost-effectiveness analysis is a type of economic analysis that compares the relative costs and effects of two or more alternatives. Cost-effectiveness analysis is often used where it is inappropriate to assign a monetary value to the measure of effect, e.g., a health effect. In these cases, results are usually stated as additional cost expended per additional health outcome achieved [26].

A typical example of a cost-effectiveness analysis can be found in ICRP Publication 55 [27]. There it was applied to the ventilation system design of a small uranium mine and the parameters were the calculated annual cost of protection, the measured annual collective dose (man·Sv) and individual dose annual average per group of workers (mSv), and the discomfort introduced by the ventilation system. The example combines the use of quantitative—cost, dose—and qualitative factors —comfort.

The cost-effectiveness ratio was then obtained graphically, plotting the corresponding values of cost of protection and collective dose for each option. The best option was established by the minimum cost-effectiveness ratio and the maximum reduction in the collective dose.

This method only enables the selection of an option that either minimizes the collective dose for a fixed protection cost or minimizes the protection cost for a specified collective dose averted.

### 11.2.3 Cost–Benefit Analysis

Cost–benefit analysis is a decision technique that implies weighing, in monetary terms, the total expected costs against the total expected benefits. It seeks to determine the best option of protection, i.e., the option with the minimum total cost, where total cost is the sum of the monetary cost of the option and the monetary value of the collective dose [19].

The compared factors are the protection cost and detriment cost associated with the radiation exposure, which includes both the health-related detriment and the non-health-related detriment. To calculate the detriment cost, you have to transform the collective dose into a monetary value using the reference value of unit collective dose, i.e., a monetary reference value of the avoided unit of exposure, generally denoted as the alpha value, which is the amount that it has been agreed to spend to avert one man·Sv of collective dose.

However, there are some problems associated with the alpha value. The first is to give a monetary value to human life, an issue so complex and subjective that never will be satisfactorily addressed. Another is to take into account the aversion that people have to the increased risk as the doses increase. If there is indeed a potential health risk associated with any level of dose, then there is a need to reduce the doses as low as reasonably achievable (ALARA). If assuming a non-threshold linear relationship between risk and exposure, the monetary value of a man-Sv can be obtained multiplying the probability of developing a health effect—fatal cancers and hereditary effects—by the monetary value of the health effect [19].

Table 11.1 shows some examples of α-values per man-mSv adopted by the regulatory authority in some countries, taken from the results of the survey

**Table 11.1** Man-Sv reference monetary values adopted by some national regulatory authorities

| Country | α-values per man-mSv (in Euro) |
|---|---|
| Finland | 15.44 EUR (20 USD)<br>77.21 EUR (100 USD) |
| Korea | 0–1 mSv: 13.13 EUR (17 USD),<br>1–5 mSv: 61.77 EUR (80 USD),<br>5–10 mSv: 270.23 EUR (350 USD),<br>$\geq$ 10 mSv: 1312.54 EUR (1700 USD) |
| The Netherlands | 453.78 EUR |
| Rumania | 570 EUR |
| Slovakia | $\leq$ 5 mSv: 33.19 EUR<br>5–15 mSv: 49.79 EUR<br>15–20 mSv: 199.16 EUR<br>20–50 mSv: 663.88 EUR |
| Sweden | 55.48-283.29 EUR |
| Switzerland | 2481.39 EUR |
| The Czech Republic | 100.39 EUR |
| The United Kingdom | 12.55-125.39 EUR depending on exposure situation |
| The USA | 154.30 EUR (200 USD) |

performed in 2009 by the Information System on Occupational Exposure (ISOE) [28]. Its use is only recommended in those countries, not required by regulation.

Some values shown in Table 11.1 depend on the annual individual dose level and some show their equivalence in USD (2012).

The Information System on Occupational Exposure (ISOE) is a network jointly sponsored by the OECD (Organization for Economic Co-operation and Development) Nuclear Energy Agency and the International Atomic Energy Agency to share dose reduction information, operational experience and information to improve the radiological protection optimization at nuclear power plants [29].

Currently, cost–benefit analysis has been mainly used in radiation safety to determine if there are enough resources allocated for safety, or if the dose reduction is worth the cost increment. In nuclear power plants, to inform important decisions—modification of installations or costly repairs.

### *11.2.4 Multiattribute Utility Analysis*

Unlike the preceding two techniques, multiattribute utility analysis allows for the assessment of all quantifiable and nonquantifiable factors in the judgmental processes to reach a decision. This technique uses a scoring scheme called a utility function for the relevant factors; with the property that if the score or the utility is the same for two options, then there is no preference for one or the other. It helps evaluate alternatives when conflicting objectives must be considered and also compare possible decisions [19, 30].

Multiattribute utility analysis is generally aimed to identify the relative importance of each factor and find out how well it does on each criterion. To use the multiattribute utility analysis, it is necessary to first recognize all factors in the decision, score each option according to our own attitude toward the factor, and tradeoff between the factors using our own personal criteria.

The first step is to specify the radiological protection factors assigning them a utility function that reflects their relative importance to the decision, and quantify the consequences of each protection option in terms of these factors. Generally the best outcome or the lowest adverse consequence for each factor is assigned a utility of 1 and the worst consequence a utility of 0.

The utility function describes how the weighted factors scores are added to arrive at an overall integrated utility for a particular option, using the user's judgments about the relative importance of each factor. If the results show the same total utility for two options, it means that none between them is preferred. The main advantage of this technique is that utility functions are not necessarily linear. Its flexibility also allows the analysis of parameters nonquantifiable in monetary terms.

### *11.2.5 Multi-Criteria Outranking Analysis*

The methods mentioned before have two critical conditions: (1) all factors must be measured in monetary terms or utility functions; and (2) tradeoffs between factors must be valid, i.e., each factor contribution must be compensatory for all other factors in the whole range of analyzed options. Therefore, when the factors are heterogeneous or can only be evaluated in a qualitative manner, it is more appropriate to use the multi-criteria outranking analysis, which is an alternative to the multiattribute utility technique [31].

In the multi-criteria outranking analysis, instead of calculating a total utility for all factors, all reasonable alternatives or factors are compared pairwise to decide if one outranks the other. It uses a "concordance index" to define the extent to which one option is preferred to—or outranks—another, and a "discordance index" to express the significance of the disadvantages when comparing one option to another.

It is also necessary to declare a concordance threshold, $C^*$, and a discordance threshold, $D^*$. For each pair of options, any option can be said to outrank or dominate another if $C_{1,2} > C^*$ and $D_{1,2} < D^*$. The strictest thresholds that can be imposed are $C^* = 1$ and $D^* = 0$. Options which outrank themselves should be possible solutions to the problem.

## 11.3 Application of Dose Limits

The dose limit principle is individual related and applicable only to planned exposure situations. This principle establishes that the total dose to any individual from all planned exposure situations, other than medical exposure of patients, should not exceed the appropriate limits specified by the ICRP [1]. These dose limits are established only for persons who are exposed to radiation at the workplace and for members of the public who are continually or frequently exposed to different radiation sources, other than natural background and individual's medical care. Dose limits do not apply to potential exposures or emergency situations, neither to existing exposure situations. However, recovery and restoration responders can be considered occupationally exposed workers and should be protected accordingly.

When doses are well below the threshold for any observable biological effect ($\sim$ 100 mSv), like in planned exposure situations, to date, assuming a linear non-threshold dose response (LNT) and based on the detriment-adjusted nominal risk coefficients, a certain risk of stochastic effects (cancer/hereditable effects) could be anticipated. Once again, it is not possible at present to unequivocally attribute a stochastic effect in an individual to radiation exposure, but the fact that health effects cannot be attributed to low radiation doses, does not mean that radiation risk cannot not be prospectively inferred for radiation protection purposes in planned exposure situations [2, 14].

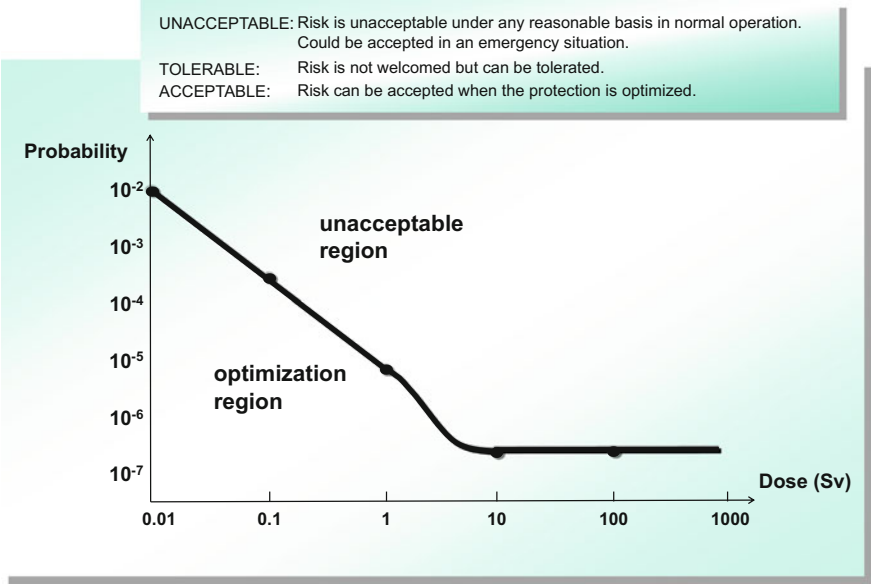


**Fig. 11.4** Criteria for risk acceptability as a dose function

The limitation principle objective is to ensure that no individual is exposed to unacceptable radiation risks in planned exposure situations. This means that measures for controlling radiation risks must ensure that no individual receives a dose greater than 100 mSv within a short period of time or, in one year, except under emergency circumstances [15].

Criteria for risk acceptability are explained graphically in Fig. 11.4. Unacceptable risk means the associated dose level that could not be reasonably accepted under normal circumstances. However, it could be accepted in unusual situations like an accident or in a given specific background, e.g., an existing exposure or in space-based activities. Tolerable risk is the associated dose level that is not welcomed, but could be reasonably tolerated if risk reduction is impracticable. Accepted risk is the dose level that is unconditionally accepted when the protection has been optimized, i.e., the dose is and has been kept well below the dose limit in the giving circumstances.

The regulatory authority determines the national dose limits taking into account the international recommendations; they apply to workers and members of the public.

Occupational exposure and public exposure dose limits for planned exposure situations [15] are summarized in Table 11.2. Dose limits for workers and members of the public are expressed in effective dose to individuals and in equivalent dose to particular individuals' organs, i.e., lens of the eye, skin and hands and feet. The annual limit of effective dose is the sum of the relevant doses from external exposure in the specified period and from the internal exposure in the same period, as a result of any intake of radionuclides via inhalation or ingestion.

**Table 11.2** Dose limits for planned exposure situations

| | Occupational | Public |
|---|---|---|
| Effective dose | 20 mSv per year averaged over five consecutive years and of 50 mSv in any single year | 1 mSv in a year. In special circumstances a higher value could be allowed in a single year, provided that the average over 5 consecutive years does not exceed 1 mSv per year |
| Equivalent dose | | |
| Lens of the eye | 20 mSv per year averaged over 5 consecutive years and of 50 mSv in a single year | 15 mSv in a year |
| Skin | 500 mSv in a year | 50 mSv in a year |
| Hands and feet | 500 mSv in a year | |

For occupational exposure of workers over the age of 18 years, the effective dose limit from all possible combined regulated sources, averaged over five consecutive years, is 20 mSv per year and should not exceed 50 mSv in any single year. There are work places or conditions where a worker is exposed in an inhomogeneous radiation field or where only parts of the body are exposed to radiation. In such cases, the annual equivalent dose for preventing the occurrence of deterministic effects should not exceed the limits to the skin, extremities (hands/feet) and lens of the eye. The equivalent dose limit for the skin is the average dose over 1 $cm^2$ of the most highly irradiated skin area.

Additional restrictions apply to occupational exposure for a female worker who has notified pregnancy or is breast-feeding. The dose to the unborn child shall not exceed 1 mSv during the remainder of pregnancy (including contributions due to mother's internal exposure). These limits can be achieved by arranging the working conditions. In the United States, the Nuclear Regulatory Commission (NRC) has established a limit of 5 mSv for pregnant women.

The occupational exposure of young workers—from 16 to 18 years—is restricted to an effective dose of 6 mSv per year.

In special and infrequent circumstances, a member of the public can receive up to 5 mSv in a single year from planned exposure, provided that the exposure is justified and not likely to occur often in his/her lifetime.

## References

1. ICRP (2007) The 2007 recommendations of the international commission on radiological protection. Ann ICRP 37, no 2–4, Apr–June 2007
2. UNSCEAR (2012) Biological mechanisms of radiation actions at low doses. A white paper to guide the Scientific Committee's future programme of work, UN, New York
3. Sources, effects and risk of ionizing radiation UNSCEAR 2012 report, report to the general assembly scientific annex a (2015) Attributing health effects to ionizing radiation exposure and inferring risks, UN, New York

4. Brenner DJ, Doll R et al (2003) Cancer risks attributable to low doses of ionizing radiation: assessing what we really know. Proc Natl Acad Sci 100(24):13761–13766
5. Feinendegen LE, Pollycove M, Sondhaus CA (2004) Responses to low doses of ionizing radiation in biological systems. Nonlinearity Biol Toxicol Med 2(3):143–171
6. Tubiana M, Aurengo A (2005) Dose-effect relationship and estimation of the carcinogenic effects of low doses of ionizing radiation. The joint report of the académie des sciences (Paris) and of the Académie Nationale de Médecine, Académie des sciences—Académie Nationale de Médecine, Paris
7. Brooks AL (2012) A History of the United States department of energy (DOE) low dose radiation research program: 1998–2008, review draft
8. Ghiassi-nejad M, Mortazavi SMJ et al (2002) Very high background radiation areas of Ramsar, Iran: preliminary biological studies. Health Phys 82(1)
9. Mortazavi SJ, Mozdarani H (2013) Non-linear phenomena in biological findings of the residents of high background radiation areas of Ramsar. Int J Radiat Res 11(1)
10. Leuraud K et al (2015) Ionising radiation and risk of death from leukaemia and lymphoma in radiation-monitored workers (INWORKS): an international cohort study. Lancet Haematol 2:276–281
11. National Institute for Occupational Safety and Health (2005) An epidemiologic study of mortality and radiation-related risk of cancer among workers at the Idaho National Engineering and Environmental Laboratory (INEEL), a U.S. Department of Energy Facility, NIOSH
12. Beyea J (2012) The scientific jigsaw puzzle: fitting the pieces of the low-level radiation debate. Bull At Sci 68(3):13–28
13. ICRP Publication 99 (2005) Low-dose extrapolation of radiation-related cancer risk. Annals of the ICRP, vol 35, no 4
14. Gonzalez AJ (2014) Clarifying the paradigm on radiation effects & safety management: UNSCEAR report on attribution of effects and inference of risks. Nucl Eng Technol 46(4)
15. International Atomic Energy (2014) Radiation protection and safety of radiation sources: international basic safety standards, IAEA safety standards series GSR part 3. IAEA, Vienna
16. UNSCEAR 2013 Report (2014) Sources, effects and risk of ionizing radiation. Scientific annex a. levels and effects of radiation exposure due to the nuclear accident, Vol I. UN, New York
17. International Atomic Energy Agency (2014) Justification of practices, including non-medical human imaging. general safety guide no GSG-5. IAEA, Vienna
18. U.S. Preventive Services Task Force (2005) Final Update summary: breast cancer: screening. U.S. preventive services task force. http://www.uspreventiveservicestaskforce.org/Page/Document/UpdateSummaryFinal/breast-cancer-screening. Accessed 2005
19. International Atomic Energy (2002) Optimization of radiation protection in the control of occupational exposure, report series no 21. IAEA, Vienna
20. NCRP Report No 120 (1994) Dose control at nuclear power plants. Bethesda, Maryland
21. International Atomic Energy (1999) Radiation protection and safety in industrial radiography, safety report series no 13. IAEA, Vienna
22. International Atomic Energy (1990) Operational radiation protection: a guide to optimization, safety series no 101. IAEA, Vienna
23. International Atomic Energy (1987) Procedures for the systematic appraisal of operational radiation protection programmes, TECDOC-430. IAEA, Vienna
24. Domenech H (1996) Utilización del método del árbol descriptivo para la elaboración de un Programa de Protección Radiológica. In: Memorias del III Congreso Regional de Seguridad Radiológica y Nuclear. Protección Radiológica en América Latina y el Caribe. Proyecto Arcal XVII/OIEA, vol 1. Cusco
25. Tague NR (2005) Quality toolbox. ASQ Quality Press, Milwaukee
26. Muenning P (2007) Cost-effectiveness analysis in health: a practical approach, 2nd edn. Jossey-Bass, San Francisco

27. ICRP Publication 55 (1990) Optimization and decision making in radiological protection. Ann ICRP 20, no 1
28. ISOE European Technical Centre (2012) ISO information sheet no 55 (2012) man-sievert monetary value survey (2012 update). ISOE European Technical Centre, Fontenay-aux-Roses
29. Nuclear Energy Agency (2015) International atomic energy, ISOE network information system on occupational exposure, NEA, IAEA. http://www.isoe-network.net/. Accessed 2015
30. Yoon PK, Hwang C-L (1995) Multiple attribute decision making: an introduction. SAGE Publications
31. Arrow KJ, Raynaud H (1986) Social choice and multicriterion decision-making. MIT Press
32. Mortazavi SJ, Mozdarani H (2013) Non-linear phenomena in biological findings of the residents of high background radiation areas of Ramsar. Int J Radiat Res 11(1)

# Chapter 12
# Occupational Radiation Protection

## 12.1 Occupationally Exposed Individuals

Occupational exposure is the radiation exposure incurred at work, i.e., planned exposure of workers at nuclear power plants and fuel cycle facilities, as well as of workers who use radiation sources, accelerators, and *X*-ray machines in medicine, scientific research, education, agriculture, and industry, etc.

Occupational exposure also includes radon in workplaces other than mines [1], and certain occupations involving the handling of naturally occurring radioactive materials (NORM) or technologically enhanced naturally occurring radioactive materials (TENORM).

At oil and gas drilling sites, for example, workers can inhale radon gas and be exposed to the alpha and gamma radiation from the decay of naturally occurred radium. Naturally occurred radionuclides can also concentrate in the mineral scales that form in pipes, storage tanks, or other extraction equipment [2]. TENORM can also be found in some building materials where coal ashes are used, in mining wastes from uranium mining, in fertilizers and fertilizer production, in scrap metals, etc. [3]. Some soils may contain residual radioactive materials from past military uses or accidents as well. In most cases, exposure to NORM and TENORM can be reduced just following safety guidance. In the United States, EPA is responsible for setting federal radiation standards for exposure to NORM and TENORM [4]. Each state has one or more programs to address both materials.

In aviation, aircrew, who repeatedly fly for years, are also considered occupationally exposed to ionizing radiation from natural sources—galactic cosmic radiation. Pilots are exposed to greater doses than cabin crew, as the passenger cabin provides more shielding than the cockpit. The average individual dose is around 3 mSv/y; however, there is considerable variation in the amount of cosmic radiation which can affect flight crews. The main variables are the flight duration and altitude,



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geographic latitude, and the solar cycle. Galactic cosmic radiation levels over the Polar Regions are about twice those over the geomagnetic equator, because of the greater amount of radiation shielding provided by the Earth's magnetic field. Spaceflights that enter the Van Allen radiation belts[1] dramatically increase levels of exposure to radiation, an item of concern for astronauts.

The Federal Aviation Administration (FAA) of the U.S. Department of Transportation is the authority which recommends the limits for aircrew—20 mSv per year averaged over 5 consecutive years, and no more than 50 mSv in a single year—for pregnant air crew, the limit is 1 mSv, with no more than 0.5 mSv per month [5].

Workers who perform recovery operations, i.e., liquidation or decontamination, after an accident and/or restoration actions in an existing exposure situation, can be regarded as occupationally exposed as well. In these cases, the work can be planned and performed in such a way that the exposure of individuals is not higher than the limits of exposure for workers [6].

## 12.2 Objectives

Occupational radiation protection requirements are specifically aimed to:

- Assure proper safety capacities, equipment, and services, commensurate to the magnitude and likelihood of occupational exposures, including:
  - engineered structures to separate physically the source from the worker—shielding, containment, interlocks, etc.;
  - ventilation systems;
  - access restriction systems and procedures;
  - operational, safety, and monitoring procedures;
  - personal protective clothing and equipment;
  - dose rate and contamination alarms, as well as equipment for dose rate monitoring, and surface and airborne contamination monitoring;
  - health surveillance and services;
- Appoint a Radiation Safety Officer (RSO) to be responsible for implementing the radiation protection program and, when appropriate, create a Radiation Safety Committee to assist the RSO in policies and technical matters;
- Designate and delimit controlled and supervised areas for preventing and confining the extent of exposures. Also, to assure that workers who are not directly related to the radiation sources receive the same level of protection than members of the public;

[1]The Van Allen belts are a collection of charged particles, gathered in place by Earth's magnetic field.

- Establish detailed local rules, including relevant investigation levels and/or operational reference levels, and indicate the steps to be taken in the event that any of such values are exceeded;
- Maintain radiation protection records to demonstrate compliance, and to allow for the review and trend analysis of occupational exposures;
- Assure the accomplishment of individual and workplace monitoring programs to verify compliance with the requirements for protection and safety;
- Make arrangements for the provision of services for personal dosimetry, environmental monitoring, and calibration of monitoring and measuring equipment, if such services cannot be performed locally;
- Ensure the adequate level of personnel education and training; and
- Promote a suitable safety culture.

## 12.3 Radiation Safety Officer (RSO)

A radiation protection, or radiation safety, officer (RPO or RSO) is a person technically competent in radiation protection matters relevant for a given type of practice, who is designated by the registrant or licensee to oversee the application of pertinent requirements established in the safety standards [7].

Under the provisions of NRC 10 CFR [8], licensees should appoint a Radiation Safety Officer to be responsible for implementing the radiation protection program. The licensee, through the radiation safety officer, shall ensure that radiation safety activities are being performed in accordance with licensee-approved procedures and regulatory requirements.

The radiation safety officer duties and authorities include the following:

- Interpret new and existing regulations, and advise the high management on technical and regulatory issues regarding the strategic direction of the radiation safety program;
- Ensure that activities involving radiation exposure are being performed in accordance with licensee-approved procedures and regulatory requirements;
- Develop or assist with the development and implementation of radiation safety policies, procedures, and local rules, as appropriate;
- Take part in the planning of activities involving significant exposures, and advise on the conditions under which work can be undertaken in controlled areas;
- Advise users of radioactive materials and radiation sources concerning the kind of materials and equipment that can be used under existing licenses and about their safe storage;
- Evaluate the effectiveness of existing radiation policies and programs, and identify radiation safety problems;
- Initiate, recommend, or provide corrective actions, stop, when necessary, unsafe operations and verify the implementation of corrective actions;

- Maintain records of all authorizations, receipts, transfers, and disposal of radioactive materials and radiation sources;
- Maintain an inventory of radioactive material and radiation sources;
- Respond to and investigate radiological incidents or accidents, and prepare and submit the required reports;
- Monitor all areas where radioactive materials or radiation sources are used on a routine and/or nonroutine basis, and maintain the corresponding records;
- Arrange for the necessary personnel dosimetry and bioassay services and maintain the corresponding records;
- Ensure the proper disposal of radioactive waste;
- Administer the radiation safety training program and ensure that all staff working with radioactive material or radiation sources is properly trained.

The radiation safety officer should have enough authority, organizational freedom, time, resources, and management prerogative to fulfill his/her duties and responsibilities [8].

In hospitals or universities, it may be also necessary to constitute a Radiation Safety Committee including all authorized users of radioactive materials and/or radiation sources, the radiation safety officer, a representative of the general management, and other members as appropriate. The Committee, among other responsibilities, assists the RSO in developing new policies and procedures; reviews and approves the uses of radioactive material within facilities; reviews radiation safety incidents, issues, and violations, and recommend corrective actions; reviews the radiation protection program to determine that all activities are being conducted in accordance with radiation safety policy, license conditions, and regulatory requirements; reviews the reports prepared by the RSO on the overall status and operation of the radiation safety program; and periodically audits the RSO.

The radiation safety officer, or radiation protection manager at nuclear power plants, should be independent of the production, operation, and maintenance groups, and advise the plant management on the effectiveness of the radiation protection program and the Radioactive Waste Management Program [9]. The radiation safety officer should also have access to all managers who have authority to establish and enforce appropriate procedures for the safe performance of work. Its responsibilities include, besides the mentioned above, the methods and procedures for implementing the radiation protection program. In United States, the power reactor health physicist is responsible for all phases of radiation protection at a nuclear reactor plant.

## 12.4 Radiation Protection Program

Facilities and activities, where radioactive materials and/or radiation sources are used, need to have in place a radiation protection program commensurate with the magnitude and likelihood of exposures. This program is to coordinate the strategies to meet the regulatory and technical requirements, and allow for its further optimization.

The radiation protection program, as part of the management system, should reflect the application of the management responsibility for radiation protection and safety through the adoption of management structures, policies, procedures, and organizational arrangements that are capable of responding to all aspects of radiation safety in normal and emergency situations. The radiation protection program may include the protection of workers, members of the public and, when appropriate, patients; and enables to evaluate whether or not the efforts for protection are reasonable and adequate. It is also important that the radiation protection program is established and managed together with other health and safety disciplines, such as industrial hygiene, industrial safety, and fire safety [10].

The first step toward the definition of a radiation protection program is to perform a prior radiological evaluation of the activity or facility, considering both normal and potential exposures. This evaluation includes a description, as precisely as necessary, of the sources of routine and reasonably foreseeable potential exposures, and a realistic estimate of the relevant doses and probabilities considering the existing conditions—shielding, containment, interlocks, ventilation, etc.—and the characteristics of neighbor buildings and facilities.

A prior radiological evaluation should also pay attention to the security of the radiation sources; the intrinsic security to be considered. The intrinsic security is the security by design, i.e., the features conceived to assure safety within the source, equipment, or facility.

A radiation protection program includes and describes, at an adequate level of detail, all of the following:

(a) The safety policy and accountability for safety at all levels, including specific responsibilities and authorities assignments;
(b) A map with the designation of areas classified as controlled and supervised including their delimitation;
(c) All the applicable constraints, reference levels, and investigation levels, if appropriate;
(d) The local safety rules, including provisions for the reception, storage, inventory, and disposal of radioactive materials and radiation sources;
(e) The applicable operational procedures and supervision of compliance procedures;
(f) The procedures for workplace monitoring including instrument specifications and calibration;
(g) The type and frequency of individual monitoring and exposure assessment, including procedures for calculation, and collection and calibration of dosimeters, when appropriate;
(h) Available systems for the registry of all relevant information related to protection and safety, including licenses and registrations, source and instrument calibrations, limits and technical conditions for source operation, effluent and radioactive waste documentation, assessment reports, individual monitoring, decisions regarding measures for occupational radiation protection and safety, incidents and accidents, etc.;

(i) The content, extension, and frequency of education and training programs for the personnel, including management and administrative staff;
(j) The approaches and procedures to periodically review and audit the program performance;
(k) Postulated events conducting to potential exposures and measures to be implemented in case they occur;
(l) The content of the emergency plan;
(m) The medical surveillance of workers;
(n) The content of the quality assurance program and compliance procedures.

## 12.5 Classification of Areas

Identify and segregate working areas consistent with radiation risk is a requirement to control radiation exposures and assure the protection of workers. Designated areas are clearly defined and classified as part of the radiation protection program, and as a result from the prior radiological evaluation. Based on operational experience and judgement two types of area may be defined—controlled areas and supervised areas.

### *12.5.1 Controlled Areas*

Controlled areas are delimited areas in which "specific protective measures or safety provisions are required to control normal exposures, prevent the spread of contamination, and prevent or limit the extent of potential exposures" [6]; ["Reproduced with permission by the IAEA"]. To determine the particular boundaries of each controlled area, it is needed to know the expected normal exposures, the likelihood of potential exposures, and the nature and extent of the required protection and safety procedures. Personnel working in controlled areas must follow local rules and work procedures to assure compliance with relevant dose limits.

Controlled areas are usually delimited by physical barriers—gates, interlocks, fences, walls, corridors, etc.—or occupy separate buildings, and are provided with an access control area at the entrance. Administrative procedures must be followed to get access, e.g., a work permit or authorization and the entry is restricted to authorized personnel only.

The standard radiation warning symbol shown in Fig. 12.1—a trefoil, as specified in ISO R 361—is displayed at the entrance, and is also used to identify specific rooms within controlled areas, accompanied by other messages that describe the nature of the radiation hazard—e.g., restricted area, radiation area, airborne radioactivity, radiologically controlled area (RCA), etc.

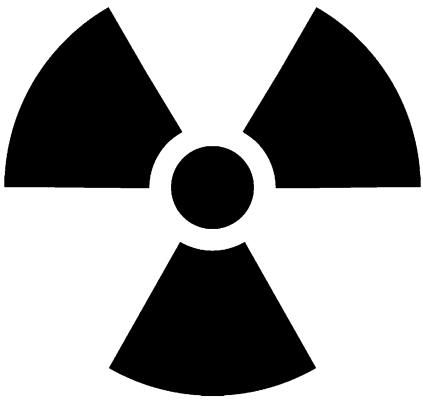

**Fig. 12.1** Basic ionizing radiation symbol

Points of access or access control areas are usually provided with personal protective clothing and equipment, individual monitoring and/or workplace monitoring, and lockers for personal clothing, as appropriate. For example, at the point of access to a controlled area with risk of external exposure, the requirement is to provide for personal dosimeters, visible and audible warning signals of radiation and, in some circumstances, an actuating device that, upon an attempt of entry, causes the level of radiation to be reduced to acceptable levels.

When working with radioactive materials that can result in contamination of the air and surfaces, the access control area is usually separated in two distinct zones—clean and unclean. Dosimeters, protective clothing and equipment, and lockers for personal clothing are located at the entrance, in the clean zone, while the equipment to monitor skin and clothing, as well as any object or substance being removed from the area, is located at the exit, i.e., at the entrance to the unclean zone. Personal decontamination facilities—wash-hand sinks, showers, etc.—and suitable storage to collect contaminated protective clothing and equipment are usually provided between both zones.

The NRC definition of a restricted area is the same as a controlled area, i.e., any area to which access is controlled for the protection of individuals from exposure to radiation and radioactive materials [11].

Posting requirements in United States include [12]:

- Radiation areas;
- High or very high radiation areas;
- Airborne radioactivity areas; and
- Areas or rooms in which radioactive material is used or stored.

A radiation area is any area with radiation levels > 0.05 mSv/h at 30 cm from the source, or from any surface through which the radiation penetrates. A high radiation area is any area with dose rates > 1 mSv/h at 30 cm from the source, or from any surface through which the ionizing radiation penetrates. A very high radiation area is any area, accessible to individuals, in which radiation levels from radiation sources

external to the body could result in an individual receiving an absorbed dose in excess of 5 Gy in 1 h at 1 m from a radiation source, or 1 m from any surface that the radiation penetrates. An airborne radioactivity area is a room, enclosure, or area in which airborne radioactive materials exist in such concentration that, during the hours an individual is present in a week without respiratory protective equipment, the individual could exceed an intake of 0.6 % of the ALI [12].

When radiation sources or radioactive materials are used in field conditions, in temporary jobsites, or are moved from place to place, appropriate posting, warning signs, temporary barriers, and direct surveillance are used to prevent unauthorized entry to the controlled area. Temporary safety barriers could be perimeter fencing and guards, portable security gates, mounting posts, mobile shielding barriers, etc. In industrial radiography or well logging applications, it may be useful as well to define a controlled area perimeter in terms of dose rate at the boundary, and specify the exposure times within this perimeter.

Individuals who do not usually work in controlled areas, but who might have access for maintenance, administrative, training, or other reasons, should be instructed in the procedures to be followed before entering. The doses these workers receive are evaluated using individual dosimeters, ratemeters, or from the results of workplace monitoring, even if they are not to be expected to receive a total dose of more than 1 mSv a year.

### *12.5.2 Supervised Areas*

Supervised areas are those not already designated as a controlled area, but where "occupational exposure conditions need to be kept under review, even though specific protective measures and safety provisions are not normally needed" [6]; ["Reproduced with permission by the IAEA"]. The purpose is "to identify parts of a workplace that should be subject to regular review of the radiological conditions to determine whether the status of the area should be changed—as a result, for example, of circumstances that were not foreseen in the prior radiological evaluation—or whether there has been some breakdown of control, either in the design features or in the procedures that operate in any adjacent controlled area" [10] ]"Reproduced with permission by the IAEA"].

Although not necessarily posted with the radiation warning symbol, supervised areas are to be clearly recognized too; it may be appropriate to make use of existing physical boundaries—doors, walls, workbenches, columns, etc. The access to these areas is usually managed using little formalities, and a frequent surveillance is required to detect any deviation from normal expected exposures. Work at supervised areas is likely to give a dose exceeding 1 mSv in a single year to any individual.

Examples of supervised areas are: liquid scintillation counting rooms and/or workstations, patient waiting areas, low-dose radiographic imaging devices operation rooms, industrial or analytical radiation-generating equipment surrounding

areas, offices and corridors adjacent to controlled areas, etc. Laboratories using little quantities of unsealed sources, laboratories for testing and processing naturally occurring radioactive materials, biomedical research and/or diagnosis laboratories using radioimmunoassay procedures, and specific areas for radiotracer studies in research and development, etc., may also be considered supervised areas since they can cause the spread of contamination.

It is important to keep in mind that is not necessary to set up a supervised area around every controlled area; using overcautious typically leads to unreasonably extent of surveillance areas, and the unnecessary measurement of negligibly levels of radiation.

## 12.6 Procedures and Record Keeping

Any approach for planning and assessing radiation protection measures should be based on a prior evaluation of the exposure situation to reasonably identify and estimate the type and magnitude of actual and potential exposures [13]. Optimization is afterwards required to reduce or avoid such exposures, establishing dose constraints and investigation levels, if appropriate, and continually improving control over radiation sources and working procedures.

Examples of better control over radiation sources are the following:

- Improving beam focusing and collimation wherever applicable;
- Minimizing scattered radiation from samples, systems, and shielding;
- Using fewer amounts of radioactive materials whenever possible;
- Replacing high energy and long-lived radionuclides with others of lesser energy and half-life;
- Using decay storage for the clearance of radioactive waste containing radioisotopes with a half-life of less than about 100 days and for biohazardous radioactive waste [14];
- Flushing radioactive systems and decontaminating to reduce the amount of radioactive materials that contributes to radiation levels in an area.

Limiting concomitant risk factors which may potentiate radiation effects—e.g., the presence of ozone in irradiation chambers or genotoxic agents in laboratories—is a way to improve working conditions for safety. The use of alternative techniques when applicable, for example, ELISA instead of RIA, or an ultrasound exam instead of radiography, is another way of reducing unnecessary exposures.

Operational procedures, local rules, and safety provisions should also assure adequate levels of protection and safety for employees, and other individuals working in controlled areas; such procedures, rules, and provisions are to be followed to avoid unexpected events.

Operational procedures are written instructions intended to document how to perform routine activities. They normally cover the risks involved, the required level of monitoring, and personal protective equipment (PPE), the potential

contingencies, and actions in case of unacceptable changes or unpredictable events. Also, the logbooks to complete, signalization, mobile or temporary shielding, estimated dose rates, and any other information required for personnel safety.

Local rules refer to a particular set of administrative rules and safety provisions governing the activities in the controlled areas; they include: the organizational structures and procedures to be followed; the values of any relevant investigation level or operational reference level, and what to do in the event that any such value is exceeded; as well as the manner of supervising any work involving occupational exposure [6].

Investigation levels and operational reference levels are fractions of the dose limit, expressed in terms of dose, dose rate, intake, airborne activity, and/or surface contamination, usually selected by management based on the practicability of reducing the specific exposure, to decide if some particular action or decision should be taken [10]. Investigation levels and operational reference levels are also recommended to assess the radiation protection program performance and with the purpose of optimization. Actions in the event of exceeding an investigation level or operational reference level are to be included in each activity's operational procedure.

Among other subjects, safety provisions include procedures for:

- Wearing, handling, and storing personal dosimeters;
- Limiting the activities that are permitted—no eating, drinking, or smoking;
- Wearing personal protective equipment;
- Posting controlled areas;
- Monitoring workplace, individuals, and environment, if appropriate;
- Collecting, storing, and disposing of radioactive waste;
- Decontaminating surfaces;
- Requesting, receiving, transporting, and delivering radioactive materials;
- Performing and recording sealed sources leakage tests;
- Performing and recording source calibration;
- Performing and recording instrument routine checking;
- Discharging radioactive effluents;
- Performing ionizing radiation-producing equipment quality control;
- Ways and means of record keeping;
- Responding to incidents and accidents;
- Minimizing radiation exposure during unusual events.

Record keeping is the making and maintaining of records to prove compliance with dose limits and legal regulations. These records also provide data for analysis of dose distribution and trends, for medical and/or legal purposes, and for validation of the optimization effectiveness.

Depending on the complexity of the facility or activity and the sources, the record keeping system should support the following registers, and include annual program audits and program content review. These are just examples:

- Authorization and renewals;
- Classification of areas and maps showing locations and surveillance points;
- Radioactive material inventory, including quantities received, where are they stored, used, transferred, and disposed of;
- Radiation generator inventory, including unique model and serial number, type —kilovoltage unit, teletherapy unit, particle accelerator, megavoltage beam— commissioning, location, intensity of the radiation emitted and technical specifications according to the type, and decommissioning;
- Sealed source inventory, including its unique identifier: model number and serial number, type and activity of the radioactive material, date when the activity was measured, reception, location, and when and where was transferred, decommissioned and disposed of. When the source is removed and returned—date and time of use, and identification of person removing and using the radiation source;
- Ordering and receiving of radioactive materials;
- Sealed source leak testing, including source unique identifier, testing procedure, and results;
- Individual monitoring, including dose records of all individuals for whom monitoring was required, the declaration of pregnancy in case of pregnant women and the dose received by the embryo/fetus;
- Workplace dose rate and, as appropriate, airborne activity monitoring, including measuring instrument and person who measured;
- Surface contamination monitoring results, including measuring instrument, location, and person who measured; also indicating if wipe tests were performed to detect removable contamination;
- Environment surveillance, including measuring instrument, location, sampling method, and person who measured;
- Effluent discharge, including measured activity, place of discharge, and quantity;
- Radioactive waste collection, handling and disposal, including classification, activity, primary radionuclides, origin, and location where is temporary stored and/or disposed of;
- Source calibration, including dosimetry and quality control when appropriate;
- Inspections and corrective measures;
- Building and equipment repairs;
- Personnel education, training, and qualification;
- Incident and unusual events reporting;
- Measuring instruments calibration and testing, including instrument unique identifier, testing procedure, and results;
- Workers' health surveillance results, including working conditions, external and internal exposure data, and accidental exposure or occupational disease.

Dose records include details of both external and internal exposure, assessed using individual dosimeters, internal dosimetry techniques, and workplace monitoring results. Because of its complexity and the fact that, very often, the implied doses are

so small that are difficult to interpret, the regulatory authority typically establishes recording levels. A recording level is the value of dose, exposure or intake above which a result should be retained or registered [10]. The Commission recommended that the recording level should be derived from the duration of the monitoring period, and an annual effective dose of no lower than 1 mSv or an annual equivalent dose of about 10 % of the relevant dose limit [15]. For an intake of a radionuclide, the recording level could be set to correspond to a committed effective dose of 1 mSv from a year's intakes [16]. Nevertheless, the minimum level of detection for the method is more often used as recording level.

It is also important to designate the person responsible for maintaining all records, as well as to clearly identify where the records are maintained.

Unintended or accidental medical exposures, the relevant physical and clinical parameters selected for treatment procedures, the information necessary for the retrospective assessment of doses to patients, the exposure of volunteers in a biomedical research, the doses received by emergency workers, the decisions made before, during and after a remediation, the results of all monitoring programs after completion of remedial actions, and the doses received by aircrew from occupational exposure to cosmic radiation, etc., should also be recorded and documented.

## 12.7 Monitoring Programs

Monitoring is "the measurement of dose or contamination for reasons related to the assessment or control of exposure to radiation or radioactive substances, and the interpretation of its results" [7] ["Reproduced with permission by the IAEA"].

Consistent with the location, two types of monitoring can be identified in occupational exposure: individual monitoring and workplace monitoring, which are explained below.

Three monitoring types can be distinguished according to the individual and workplace monitoring purpose. They are [10] ["Reproduced with permission by the IAEA"]:

(1) "Routine monitoring, designed to demonstrate on a periodical base that working conditions remain satisfactory and that the corresponding regulatory requirements are met;
(2) Task-related monitoring, intended to support immediate decisions on the operation management and protection optimization; and
(3) Special monitoring, designed to investigate a situation for which insufficient information is available, for example, at the commissioning stage of new facilities, following major modifications to facilities or procedures, or when operations are being carried out under abnormal circumstances such as an accident."

### 12.7.1 Individual Monitoring and Exposure Assessment

Since occupational exposure involves the external and internal exposure components, individual monitoring means the "external exposure measurement using dosimeters worn by individuals, or the measurement of quantities of radioactive material in or on, or taken into, the individual's body, or the measurement of radioactivity in samples of material excreted from the individual's body" [6] ["Reproduced with permission by the IAEA"].

As stated by 10 CFR Part 20 [12], any worker who is normally employed in a controlled area or who occasionally works in a controlled area for maintenance, training or other reasons, and whose dose is likely to exceed 10 % of the corresponding dose limit, has to be subject to individual monitoring. The use of individual monitoring devices for external exposure is then required for individuals entering a high or very high radiation area; for adults likely to receive an annual effective dose of 5 mSv in the course of their normal work, or an annual equivalent dose of 1.5 mSv to the lens of the eye, or of 5 mSv to the skin or any extremity; for minors likely to receive a committed effective dose in excess of 1 mSv in 1 year; and for declared pregnant women likely to receive, during the entire pregnancy, a committed effective dose in excess of 1 mSv [12].

Dosimeters used to monitor the dose equivalent from external exposure—i.e., the personal dose equivalent, $H_p(10)$, since the absorbed dose cannot be measured directly—are suitable for the radiation type and energy; and, where feasible and practicable, their range covers the maximum potential exposure. In most cases, it is common to wear multiple dosimeters, e.g., film bag and thermoluminescent dosimeters (TLD) for photon/beta radiation, track detectors and albedo dosimeters for neutrons, pocked chamber direct reading dosimeters, personnel accident dosimeters, etc. Personal dosimeters are worn at a body position, which is representative of body exposure. TLD dosimeters and track detectors worn on fingers, wrist, and forearms are used to measure the skin or extremity personal dose equivalent, $H_p(0.07)$.

If individual monitoring "is inappropriate, inadequate or not feasible, then it is required to assess the occupational exposure based on the results of workplace monitoring and on information on the location and duration of exposure" [6]; ["Reproduced with permission by the IAEA"]. Occupational exposure may also be assessed on this basis when the worker is regularly employed in a supervised area or when he/she enters a controlled area only occasionally.

Two conditions are required for the assessment based on the workplace monitoring to be applicable. First, the equipment and measurement procedures for workplace monitoring should have demonstrated their reliability. Second, the expected doses must be relatively constant [17]. Areas where *X*-ray inspection or *X*-ray diffraction machines are used are examples of workplaces where it is feasible to routinely assess occupational doses without individual monitoring. Most of these machines are totally enclosed and locally shielded and, in normal operation, they usually produce a dose rate lower than 2 μSv/h outside the *X*-ray apparatus.

Surveys of dose rates and estimates of occupancy times can then be used to obtain the expected doses received by workers.

Monitoring for internal exposure is required for adults likely to receive in 1 year an intake in excess of 10 % of the applicable ALIs for ingestion and inhalation, or if the committed effective dose is likely to exceed 0.5 mSv for any occupationally exposed minor, or declared pregnant woman [12].

The need for internal exposure monitoring—individual and area—depends on the amount of radioactive material present and the radionuclide(s) involved, the physical and chemical form of the radioactive material, the type of containment used, and the operations performed [16]. A routine monitoring program for internal exposure is recommended when handling large amounts of gaseous and volatile materials—e.g., in heavy water reactors, reactor maintenance, plutonium and other transuranic elements processing, thorium and uranium ores mining, milling and processing, production of radionuclides, etc.—or in special circumstances such as an accident, incident, or other unusual occurrence [18].

Internal exposure monitoring is also required when the use of engineered controls (e.g., forced ventilation, filtration, etc.) to reduce the concentrations of radioactive material in the air is not practical and it is necessary to limit the intakes using respiratory protection equipment, controlling access, and limiting the exposure times [12].

The estimation of intakes for individual monitoring is based on measurements performed either in vivo (whole body counting, particular organ counting), which usually require well shielded and precise calibrated installations, or in vitro (analysis of excreta like urine, feces, sweat, and of body fluids like blood, breath, or saliva), which depends to a greater extent on biokinetic models to predict the radionuclides behavior within the body. Both methods strongly depend on the time at which the monitoring is performed and on the analytical sensitivity—i.e., the smallest amount of activity that can be accurately measured—as well [19].

Direct measurements (in vivo*)* are usually performed on photon emitters that can be measured through the body, or on insoluble radionuclides within the lungs. Beta emitters, alpha emitters, or photon of very low energy, are measured using bioassay techniques (in vitro). Exhaled air measurements are also used for some radionuclides such $^{226}$Ra and $^{228}$Th which progeny includes gases that may be exhaled—radon, thoron.

When workers are exposed to internal contamination and individual monitoring for internal exposure is inappropriate, inadequate, or not feasible, individual doses due to the intakes of radionuclides can also be estimated on the basis of the airborne activity concentration and the expected duration of exposures. Results of workplace monitoring—dose rate and levels of surface contamination and airborne contamination— including personal air sampling are then needed.

### 12.7.2 Workplace Monitoring

Workplace monitoring is "the monitoring using measurements made in the working environment" [7] ["Reproduced with permission by the IAEA"]. Workplace monitoring is performed to confirm that working conditions are satisfactory and that expected levels of occupational exposures are met. Data from workplace monitoring can be used to identify changes in radiological conditions, to evaluate operational and safety procedures, and to corroborate the best practices are implemented. It also serves as a basis for reviewing the classification of areas. Workplace monitoring is used as well to estimate doses received by workers when individual monitoring is inadequate or not practicable.

As with individual monitoring, workplace monitoring involves measurement, assessment, and interpretation, as well as compliance with quality assurance requirements to ensure that procedures are established and followed correctly, and records are promptly made and correctly maintained. It includes the following:

- Ambient dose equivalent, airborne radioactivity, and surface contamination measurements in selected points, considering radiation types and radioactive materials forms;
- Selection, purchase, maintenance, and calibration of radiation monitoring instruments, tools, and personal protective devices;
- Appropriate operational state updated records; and
- Availability of the instruments, tools, and personal protective devices for normal and emergency conditions.

Resources for and frequency of workplace monitoring depend on the prior radiological evaluation—expected normal exposures and magnitude and likelihood of potential exposures—and the actual levels being measured. Workplace monitoring planned instruments may include: fixed area radiation monitors, portable dose rate meters, continuous airborne contamination monitors, area air samplers, personal air samplers, portable surface contamination monitors, fixed contamination monitors, and laboratory equipment—e.g., liquid scintillation counters, gamma spectroscopy, etc.—to properly identify radionuclides and for measuring smears, filters, liquid samples, etc.

For example, a fixed radiation monitor with a warning system or alarm is commonly positioned in an area where an unpredictable radiation level increase is likely to occur, and a fixed contamination monitor is normally located at a controlled area exit to prevent workers from spreading contamination. Also, a portable surface contamination monitor is usually employed to detect inadvertent spills and/or containment failures, and an airborne radioactivity monitor, when handling gaseous or volatile radioactive materials.

Monitoring frequency and instrument type are determined by the likelihood of changes in the radiological conditions of controlled and supervised areas, the radiation type and energy to be measured, and the minimum detectable concentration requirements [20].

## 12.8 Calibration

Monitoring instruments calibration is important to ensure their measuring operational state and the quality and accuracy of individual and workplace monitoring. Calibration of radiation protection monitoring instruments is performed before first use, periodically—at least once a year according to regulations—and following a major repair or maintenance [21].

Calibration is essentially a process of comparison; it is the process of determining, under specified conditions, the relationship between the readings obtained by a measuring instrument or system and the known value and uncertainty of the standard. The Radiation Physics Division, part of the Physical Measurement Laboratory at the National Institute of Standards and Technology (NIST), develops, maintains, and disseminates the measurement standards for ionizing radiations and radioactivity in the United States [22].

Calibrations of dosimeters and/or instruments for workplace monitoring may be performed by the facilities themselves or by specialized, independent calibration services. In any case, laboratories conducting calibration of radiation detection instruments and dosimeters are to be accredited by the National Voluntary Laboratory Accreditation Program (NVLAP) [23], a federal program run by NIST, and receive the corresponding general license from NRC or the Agreement State.

Measurements results for calibration are to be traceable to primary[2] or secondary[3] standards. Metrological traceability permits comparison of measurements, whether the result is compared to the previous result in the same laboratory, a measurement result a year ago, or to the result of a measurement performed anywhere else in the world. Test methods, result interpretation, and requirements for instrument calibration could be found in various ANSI and ASTM Standards, as well as in different IEC Standards, and in NCRP Publication No. 112 [24].

The result of a calibration is recorded in a document sometimes called a calibration certificate or a calibration report. Records of radiation survey instrument calibrations are to be retained for 3 years following the date the record was created.

## 12.9 Personnel Education, Qualification, and Training

Considering the critical role that human actions play on safety, a major component of the radiation protection program is, and should be, building competence in protection and safety. Competence means the "ability to apply knowledge, skills

[2]It is a standard with the highest metrological qualities in a specified field. Primary standards are maintained at national laboratories that (a) perform research for the purposes of metrology and (b) participate in recognized international intercomparison of primary standards laboratories—NIST.

[3]It is a standard whose value is fixed by direct comparison with a primary standard and which is accompanied by a certificate that documents this traceability.

and attitudes so as to perform a job in an effective and efficient manner" [25] ["Reproduced with permission by the IAEA"].

Most regulatory authorities have in place provisions for education, qualification, and training of workers that have any function and/or responsibility with regards to safety. Some examples are the requirements for education, qualification, and experience for authorized users, medical physicists or authorized nuclear pharmacists [26] and the initial licensing, including written examinations and operating tests, for commercial reactor operator's [27].

Radiologic technologists, the medical personnel who perform diagnostic imaging examinations and administer radiation therapy treatments, are usually licensed or certified by States. Currently, most of the states have licensing laws covering the practice of Radiologic technology. Furthermore, the American Registry of Radiologic Technologists (ARRT) is the primary source for verification of education, training, work experience, license, etc., of radiologic technologists [28].

Individuals who work in different medical applications—e.g., imaging and localization studies using unsealed radioactive sources; uptake, dilution, and excretion of radiopharmaceuticals; oral administration of sodium iodide $^{131}$I, use of manual brachytherapy sources, etc.—are certified by the respective national board or specialty. Specialty Boards Certification recognized by NRC includes [29]:

- The American Board of Health Physics
- The American Board of Science in Nuclear Medicine
- The American Board of Radiology
- The Canadian College of Physicists in Medicine
- The Board of Pharmaceutical Specialties
- The American Board of Nuclear Medicine
- The Certification Board of Nuclear Cardiology
- The American Osteopathic Board of Radiology

Qualification requirements usually entail a minimum educational level and specific training and work experience. Generally, they include the successful completion of a number of hours of specific classroom and laboratory training; a number of years of experience, or supervised experience, working with similar practices or procedures; and on-the-job practical training in regulatory issues, and operating and emergency procedures associated with the specific task, activity, or facility.

In academic and research applications, for instance, the authorized user should have as a minimum a college degree at the bachelor level or equivalent training and experience in physical, chemical, biological sciences, or engineering [30], while in industrial radiography, a radiographer should have received training in fundamentals of radiation safety (characteristics of radiation, units, measuring instruments, hazards, etc.), in addition to a minimum of 2 months of on-the-job training, and is certified by a certifying entity that he/she has met established radiation safety, testing, and experience criteria [31]. To facilitate the certification of industrial

radiographers, 10 states and the American Society for Nondestructive Testing, Inc. (ASNT) have nationally recognized certification programs [32].

The initial training could be received from accredited training centers or courses in the form of formal education, lectures, workshops, tutorials, seminars, or practical exercises—case studies, simulation exercises, technical visits, etc.—and/or on-the-job training if it has an appropriate program and duration. Training normally has a final written exam to assess its successful completion and to keep records of the education provided.

Training subjects may vary, but the following basic topics are usually required:

- Physics of radiation protection—radioactivity and radioactive decay, interaction of radiation with matter, units and quantities, and radiation detection principles and instruments;
- Biological effects of radiation;
- Basic principles of radiation protection and principal elements of a radiation protection program for occupational, medical and public exposure;
- Relevant legislation and regulations governing radiation safety;
- The concept of safety culture;
- Decommissioning and radioactive waste management, including safety relating to spent and disused radiation sources;
- Transportation of radioactive materials;
- Emergency planning and preparedness.

After initial training, periodic short courses, online refresher courses, seminars, and practical drills on emergency planning and preparedness, are annually offered to update training and continually maintain the necessary level of competence. Apart from specific topics, the following can be addressed during retraining:

- The system of radiation protection and its conceptual framework;
- Main regulations concerning specific practices or applications;
- Measurement instruments and their practical use;
- Lesson learned from incidents and accidents.

Since it is important that all persons associated with radiation in a facility or activity are properly trained and qualified, training programs are typically divided into different categories depending on individuals' level of education and level of safety responsibilities.

Training programs are usually planned for: personnel working directly with radiation, qualified operators, health professionals, emergency response personnel, managers, and personnel with a low potential for exposure [25, 33]. The content and level of training are different for each category.

For workers, qualified operators and health professionals, training content significantly vary according to the application. Although most of qualified and health professionals usually keep up to date with developments through attendance to formal courses, conferences, workshops, seminars, etc., radiation safety training programs are still planned to ensure they understand the risks associated with their

occupational exposure, the rules and procedures to follow on the work, the use of measuring instruments and interpretation of results, the use of safety devices, controls, and warning systems, and the actions to protect themselves and others in case on emergencies. General workers are to be periodically trained on the practical aspects of the radiation protection program, emergency procedures, and the safe use of radiation sources.

Unlike individuals who work directly with radiation and need a broader and deeper training, people who frequent any area where radiation sources are used or stored, or who work in the vicinity of radiation sources, and/or share their lab space with individuals that use radioactive materials, typically receive radiation safety awareness training. This kind of training provides them with basic radiation safety information about how to recognize the risks involved with radiation, especially the recognition of warning signs and signals, and the basic procedures.

Managers responsible for overseeing the use of radiation sources are also informed on a regular basis about the principal elements of the system of radiation protection, radiation risk management, the concept and principles of safety culture, and the regulations governing radiation protection [33].

In addition to qualification and training, other factors that need to be considered when judging competence are self-control, dependability, and responsibility. These factors may be evaluated through individual alertness, signs of emotional distress, fatigue or specific illness that could impair the ability to safely and competently perform his/her duties; the use of prescription and over-the-counter medications that could cause certain level of impairment is to be taken into account as well. General health and fitness should be certified by a physician.

Other recommended individual attributes include communication skills as to discuss safety issues, leadership skills as to start urgent actions, analytical skills as to assess radiation hazards and interpret results from monitoring, and skills to use the corresponding equipment and controls [25].

## 12.10 Conditions of Service

The requirements and recommendations on radiation protection have stated for many years now that conditions of service of workers should be independent of their occupational exposure. Furthermore, one of the requirements of occupational exposure is that "special compensatory arrangements or preferential treatment with respect to salary or special insurance coverage, working hours, length of vacation, additional holidays or retirement benefits shall neither be granted nor be used as substitutes for measures for protection and safety in accordance with the requirements" [6] ["Reproduced with permission by the IAEA"].

Another concern has been the exposure of pregnant workers and young people. Special arrangements should be made for female workers and for persons less than 18 years of age undergoing training.

Pregnancy or breast-feeding is not a reason to exclude a female worker from work, but a woman who performs her job in controlled or supervised areas, or undertake emergency duties, should declare their pregnancy as soon as suspected. When she notifies her suspected pregnancy, or if she is breast-feeding, the working conditions can be adapted so as to ensure that the embryo or fetus or the breastfed infant is afforded the same level of protection as for members of the public.

There may also be circumstances where, for health reasons or when medically recommended, an employee cannot be temporarily exposed. If necessary, arrangements like an alternative setting or different tasks can also be made for him/her to continue working.

No person under the age of 16 years should be subject to occupational exposure. Persons between 16 and 18 years of age can have access to a controlled area only under supervision, and only for training or study purposes. Condition should be such as to assure them the following exposure:

| | |
|---|---|
| Annual effective dose | 6 mSv |
| Equivalent dose to the lens of the eye in a year | 20 mSv |
| Equivalent dose to the extremities (hands and feet) or the skin in a year | 150 mSv |

## 12.11 Quality Assurance

Quality assurance is the "function of the management system[4] which provides confidence that specified requirements are fulfilled. It comprises all planned and systematic actions necessary to provide adequate confidence that an item, process, or service will satisfy given requirements for quality, for example, those specified in the license" [7] ["Reproduced with permission by the IAEA"].

As learned before, one of the main objectives of the management system is to ensure that quality requirements are not considered separately from safety requirements. By bringing together in a coherent manner all the interrelated or interacting elements that establishes policies and objectives, the management system integrates the radiation protection program with the quality assurance program and the rest of the requirements, while putting safety in first place, i.e., safety override all other demands.

The management system reflects and includes the initial concept of quality control—controlling the quality of products—and its evolution through quality assurance—the system to ensure the quality of products—and quality management—the system to manage quality [34]. The goal of quality is then to ensure that all

[4]See Management system for safety in Chap. 10.

necessary actions to control occupational exposure have been taken, so that the entire system is within specifications, under a wide range of conditions of operation.

The quality assurance part of the system should pay attention, among others, to the following:

- That documented safety policies and procedures are implemented and in compliance with requirements;
- That quality documents and records are maintained, audited, and harmonized with safety procedures. This include calibration of instruments, equipment, and sources; testing of instruments, sources, and engineered controls; and validation of calculation software, and computer information systems for individual monitoring, workplace monitoring, source inventory, training, etc.;
- That competence, awareness, education, and training plans are available and documented;
- That those resources such as workspace, equipment, support services, information and communication technology, and transport facilities, for normal and emergency situations, are provided;
- That the effectiveness of safety is assessed, reviewed, and reported on a regular basis through workspace inspections or observations; reviews, analysis and trending of important performance and safety data; reviews of new corrective action reports; and internal audits;
- That reference operational levels, investigation levels, and recording levels are established, used and reviewed, as appropriate;
- That any modification to facilities, equipment, or sources is reviewed and approved by the Radiation Safety Officer or the Radiation Safety Committee.

## 12.12 Medical Surveillance

Medical surveillance of workers exposed to radiation should be based on the general principles of occupational safety and health, which aims to assess the initial and continuing compatibility between the health of workers and working conditions, and to provide a baseline of information useful in the case of accidental exposure or occupational disease [35].

Medical surveillance of workers exposed to radiation should also be associated to the specific job and health conditions required to effectively perform particular responsibilities. For this to be accomplished, it is important that occupational medicine physicians are familiar with the specific work processes, job requirements, and existing hazards. Occupational medicine physicians, working together with radiation safety officers, should schedule medical examinations and assessments of the workplace safety and hygiene conditions on a regular basis, as well as undertake first aid measures in the event of an emergency.

Occupational health services usually include a preemployment (or preplacement) examination to decide if the worker is fit to perform his/her job, and periodical

examinations to determine the continuing fitness. The frequency of periodical examinations depends on local regulations, the specific application the worker is involved with, and other circumstances such as age, gender, etc. In addition, workers may receive a medical examination to return to work after an abnormal exposure, and before the employment terminates.

Three situations may require special medical examination. These are [36] ["Reproduced with permission by the IAEA"]:

- "Where the work involves potential exposure to airborne radioactive material and it is necessary to assess an individual's fitness to wear protective respiratory equipment in some areas, verification of lung function is essential;
- Where the work involves potential skin contamination and it is necessary to assess an individual's skin condition for disease or damage that could either be exacerbated by contamination, accelerate absorption or preclude the wearing of necessary personal protective clothing; and
- Where the work is such that employees with psychological disorders may be a hazard to either themselves and/or their colleagues."

Medical care is also necessary in case of workers overexposure, to treat radiation injuries, and to decontaminate personnel. If there was an overexposure, physicians closely monitor people for the development of the various syndromes and treat the symptoms as they arise. Radiation burns can be treated using the same therapeutic measures applied to thermal burns, but may require surgical excisions and reconstructions [37]. In case of uptake of a radioactive contaminant, the general principles are to reduce absorption and increase excretion [38]. Certain medications could be used to decrease the absorption of radionuclides.

Although skin decontamination can be easily achieved using large amounts of soap and warm—not hot—water, if the dose received is likely to be 2 Gy or greater, a visible erythema will appear very soon after the exposure and the treatment should be handled by a doctor; all wounds and/or burns should be decontaminated only by a physician.

## References

1. International Atomic Energy (2003) Radiation protection against radon in workplaces other than mine, Safety report series no. 33. IAEA, Vienna
2. OGP Report No. 412 (2008) Guidelines for the management of naturally occurring radioactive material (NORM) in the oil & gas industry. International Association of Oil and Gas Producers
3. TENORM.com (2013) Technologically enhanced naturally occurring radioactive materials. Industrial Sectors with TENORM, NORM Solutions LLC. http://www.tenorm.com/sectors.htm
4. EPA U.S. Environmental Protection Agency (2015) Naturally-occurring radiation. TENORM. laws and regulations. U.S. Environmental Protection Agency. http://www.epa.gov/radiation/tenorm/regs.html

5. U.S. Department of Transportation, Federal Aviation Administration (2014) Advisory Circular AC 120-61B, In-Flight Radiation Exposure, John S. Duncan, Director, Flight Standards Service
6. International Atomic Energy (2014) Radiation protection and safety of radiation sources: international basic safety standards, IAEA safety standards series GSR part 3. IAEA, Vienna
7. International Atomic Energy (2007) IAEA safety glossary, terminology used in nuclear safety and radiation protection—2007 edition. IAEA, Vienna
8. NRC Regulations Title 10 (2008) Code of federal regulations, Part 35, General domestic licenses for byproduct material, Oct. 1, 2007 and July 23 2008
9. International Atomic Energy (2002) Radiation protection and radioactive waste management in the operation of nuclear power plants, Safety guide NS-G-2.7. IAEA, Vienna
10. International Atomic Energy (1999) Occupational radiation protection, safety guide no. RS-G-1.1. IAEA, Vienna
11. U.S. NRC (2015) Glossary, United States Nuclear Regulatory Commission. http://www.nrc.gov
12. NRC Regulations Title 10 (2013) Code of federal regulations, part 20, Standards for protection against radiation
13. ICRP Publication 103 (2007) The 2007 recommendations of the international commission on radiological protection. Ann ICRP 37(2–4)
14. International Atomic Energy (2006) Storage of radioactive waste, safety guide WS-G-6.1. IAEA, Vienna
15. ICRP Publication 75 (1997) General principles for the radiation protection of workers. Ann ICRP 27(1)
16. International Atomic Energy, International Labour Office (1999) Safety guide No. RS-G-1.2. Assessment of occupational exposure due to intakes of radionuclides. IAEA, Vienna
17. International Atomic Energy, International Labour Office (1999) Assessment of occupational exposure due to external sources of radiation, Safety guide no. RS-G-1.3. IAEA, Vienna
18. ICRP Publication 78 (1997) Individual monitoring for internal exposure of workers. Ann ICRP 27(3–4)
19. International Atomic Energy (2004) Methods for assessing occupational radiation doses due to intakes of radionuclides, Safety report series no. 37. IAEA, Vienna
20. ANSI/HPS N13.49-2001 (2001) Performance and documentation of radiological surveys. Health Physics Society, McLean
21. International Atomic Energy (2000) Calibration of radiation protection monitoring instruments, Safety report series no. 16. IAEA, Vienna
22. NIST, Physical measurement laboratory, Physical reference data, NIST. http://www.nist.gov/pml/data/
23. National Institute of Standards (2015) National voluntary laboratory accreditation program (NVLAP). National Institute of Standards, U.S. Department of Commerce. http://www.nist.gov/nvlap/
24. NCRP Report No. 112 (1991) Calibration of survey instruments used in radiation protection for assessment of ionizing radiation fields and radioactive surface contamination. Bethesda, Maryland
25. International Atomic Energy (2001) Building competence in radiation protection and safe use of radiation sources, Safety standards series no. RS-G-1.4. IAEA, Vienna
26. U.S. NRC (2015) Part 35—medical use of byproduct material. United States Nuclear Regulatory Commission. http://www.nrc.gov/reading-rm/doc-collections/cfr/part035/
27. U.S. NRC (2015) Operator licensing. United States Nuclear Regulatory Commision. http://www.nrc.gov/reactors/operator-licensing.html
28. ARRT (2015) State licensing. The American Registry of Radiologic Technologists®. https://www.arrt.org/state-licensing/
29. U.S. NRC (2015) Specialty board(s) certification recognized by NRC under 10 CFR part 35. United states nuclear Regulatory Commission. http://www.nrc.gov/materials/miau/med-use-toolkit/spec-board-cert.html

30. NUREG-1556 Vol. 7 (1999) Consolidated guidance about materials licenses. Program-specific guidance about academic, research and development, and other licenses of limited scope. U.S. Nuclear Regulatory Commission. Office of Nuclear Material Safety and Safeguards, Washington DC
31. NUREG-1556 Vol. 2 (1998) Consolidated guidance about materials licenses. program-specific guidance about industrial radiography licenses. U.S. Nuclear Regulatory Commission Office of Nuclear Material Safety and Safeguards, Washington, DC
32. Conference of Radiation Control Program Directors (2015) Industrial radiography certification. In: Conference of radiation control program directors. A partnership dedicated to radiation protection http://www.crcpd.org/IR.aspx
33. International Atomic Energy (2001) Training in radiation protection and the safe use of radiation sources, Safety reports series no. 20. IAEA, Vienna
34. International Atomic Energy (2006) The management system for facilities and activities, Safety requirements no. GS-R-3. IAEA, Vienna
35. ILO (1998) Occupational safety and health series 72. Technical and ethical guidelines for workers health surveillance. ILO, Geneva
36. International Atomic Energy (2004) Health effects and medical surveillance, practical radiation technical manual 3 (Rev. 1). IAEA, Vienna
37. Ryan JL (2012) Ionizing radiation: the good, the bad, and the ugly. J Invest Dermatol 132:985–993
38. U.S. Department of Health and Human Services (2006) Food and drug administration, Center for drug evaluation and research (CDER), Guidance for industry. internal radioactive contamination—development of decorporation agents, Medical Clinical

# Chapter 13
# Public Radiation Protection

Protection of the public is also an integral part of the radiation safety program. When managing radioactive waste, radioactive discharges, or transporting radioactive sources, it is important to consider its impact on occupationally exposed workers, and also the impact on members of the public who may occasionally come into contact with the waste, release, or source. Think, for example, of a radioactively contaminated scrap in a landfill and the people who may have access to it.

The general public is normally exposed to various sources of ionizing radiation of different origin. These sources range from natural sources—naturally occurring radioactive materials, such as uranium, thorium, and radium; radon in air; cosmic radiation; and internal radiation—to man-made radiation sources—like the ones used in medicine, industry, agriculture, research, nuclear electricity generation, and consumer products. The public can also receive some exposure from fallout from nuclear weapons testing, unplanned satellite's reentries and radiation accidents like Chernobyl.

Contributions of all sources to the exposure of United States population in 2009 are shown in Fig. 13.1. The data for the graphic is from the NRC [1] ("Courtesy of United States Nuclear Regulatory Commission").

As noted from Fig. 13.1, exposure to natural sources of radiation—cosmic, radon and thoron, terrestrial, and internal—is almost the most significant part of the total public's exposure to radiation; we can thank existing radiation safety regulations for that matter.



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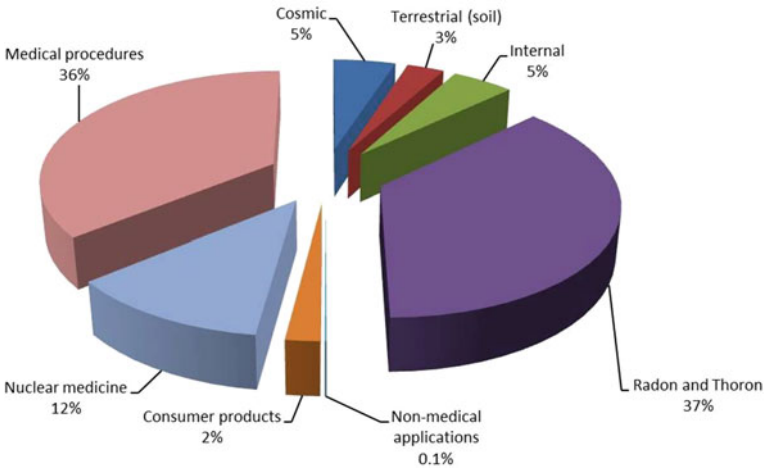


**Fig. 13.1** Public's exposure to all sources of radiation in the United States

## 13.1 Natural Sources of Radiation

Natural background—the radiation that comes from natural sources—is the baseline at which all man-made exposures are added and, against which, these exposures should be compared. Levels of natural radiation can vary greatly from one location to another depending on the site altitude above the sea level—cosmic radiation—and mineral composition of rocks and soils — terrestrial radiation.

The world population-weighted average annual effective dose due to cosmic radiation adjusted for altitude is 0.34 mSv, while the same value at sea level is 0.27 mSv, approximately 1.25 times lesser [2]. High natural radiation background due to cosmic radiation is found in high-altitude places such as La Paz, Bolivia: 2.02 mSv at 3900 m [3]; Mount Lorne, Canada: 0.84 mSv at 2000 m [4]; and Denver, United States: 0.54 mSv at 1610 m [3].

Terrestrial radiation contribution to public exposure largely comes from natural radionuclides contained in soils, rocks, plants, water—rain water, rivers, lakes, sea—and living organisms. Radionuclide concentrations in soils are basically determined by its concentration in the source rocks. Thus, there is a large variation in natural radionuclides concentration around the world since it depends on the local geology and geography. Although there are exceptions, higher radiation levels are associated with igneous rocks, such as granite, while lower levels are associated with sedimentary rocks [5].

Natural radionuclides are classified in primordial and cosmogenic. Primordial radionuclides are those that have survived since the time the elements were formed. They include several dozen naturally occurring radionuclides with half-lives of at least $10^9$ years—the estimated age of earth is $4.5 \times 10^9$ years—and are usually divided in two groups: radionuclides that occur singly (non-series) and decay directly to a stable nuclide (e.g., $^{40}K$ and $^{87}Rb$), and radionuclides that occur in

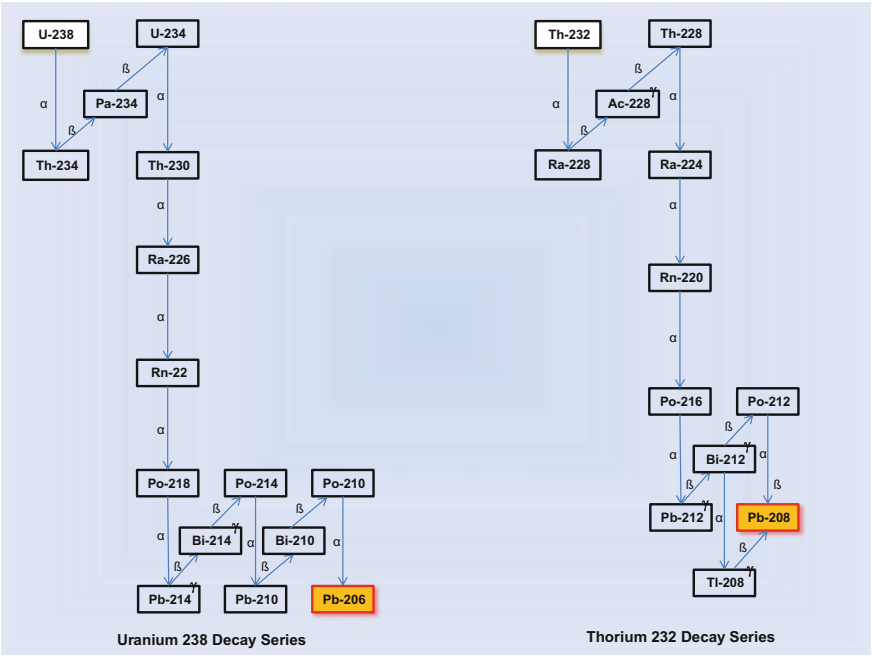


**Fig. 13.2** Uranium and thorium decay series

decay chains (series) and decay to a stable isotope of lead through a sequence of radionuclides ($^{238}$U, $^{235}$U, and $^{232}$Th). These are actually part of a whole family of radionuclides, where one decays to the next.

Figure 13.2 shows the $^{238}$U and the $^{232}$Th families as illustrations. The half-live of $^{238}$U is $4.468 \times 10^9$ years; the half-live of $^{232}$Th is $1.4 \times 10^{10}$ years. Alpha decay of radionuclides in Fig. 13.2 is indicated as $\alpha$, and $\beta$ indicates beta decay. Significant gamma emitters are indicated with a $\gamma$, and lead stable isotopes are indicated in orange.

Singly occurring primordial radionuclides such as $^{40}$K and $^{87}$Rb are inherently part of the human body. $^{40}$K is the major source of natural radioactivity in foods and water and, since its activity concentration is an order of magnitude higher than that of $^{238}$U or $^{232}$Th, it contributes to about 40 % of the exposure humans receive from natural radiation [5]. $^{87}$Rb contributes only a few percent.

External exposure from terrestrial radiation, derived from activity concentration of $^{226}$Ra ($^{238}$U), $^{232}$Th and $^{40}$K in soil, has an annual average of 0.48 mSv, ranging from 0.3 to 1 mSv [5]. Internal exposure, arising from inhalation of radon and ingestion of $^{40}$K and radionuclides from uranium and thorium series present in foods and drinking water, has an annual average of 1.26 mSv, ranging from 0.2-10 mSv, from inhalation exposure, and 0.29 mSv from ingestion exposure, depending on radionuclides composition in food and drinking water [2].

Examples of high natural terrestrial radiation background are: the monazite[1] sands of Guarapari, in the Espirito Santo State, in Brazil; the Yangjiang County in the south of China, where the sand in the region has eroded from hills containing monazite; Kerala, also a monazite-bearing coastal region in southwest India; and Ramsar, a northern coastal city in Iran with over 50 sulfurous hot springs that contain enhanced $^{226}$Ra concentrations [2, 6].

Cosmogenic radionuclides are forming continuously by the interaction of cosmic rays particles with matter. They arise from the collision of highly energetic cosmic ray particles with the $O_2$, $N_2$, and other air components at a constant rate, and are brought to the earth surface by rain water. Some of the most important cosmogenic radionuclides are $^{14}$C, $^{3}$H, $^{7}$Be, $^{22}$Na, $^{36}$Cl, and $^{38}$S. Cosmogenic radionuclides have shorter lives than primordial radionuclides—$^{14}$C half-life is 5730 years, $^{3}$H half-life is 12.32 years, and $^{36}$Cl is $3.01 \times 10^5$ years—and contribute little to population radiation doses; however, as most of primordial radionuclides, are useful for dating geologic materials and matter which was once living;

## 13.2 Man-Made Radiation Sources

Besides being exposed to naturally occurring radionuclides and cosmic radiation, general population is also exposed to other man-made radiation sources associated with technological advancements, including jet flights, enhanced sources of naturally occurring radioactive material, consumer products, and medical procedures—radiodiagnostic, nuclear medicine, and radiotherapy. Members of the public are exposed in a lower degree due to industrial, academic, and research applications of ionizing radiation, including the disposal of spent fuel and radioactive waste, and the transportation of radioactive materials.

The general population is also exposed to the global fallout resulting from nuclear testing occurred between 1945 and 1980, the activity released by nuclear satellites burnt-up in the atmosphere, and the activity released by accidents at nuclear power plants like Chernobyl and Fukushima. Some amounts of long-lived artificial radionuclides such as $^{90}$Sr and $^{137}$Cs have been added to the background radiation inventory as a result of such events.

Distribution of the average annual per caput dose to the global population according to the data reported by the United Nations Scientific Committee on the Effects of Atomic Radiation (UNSCEAR) [2] is shown in Fig. 13.3.

Many radionuclides released to the atmosphere from nuclear testing, were brought to the ground by the way of the so-called global fallout during the late 1950s and early 1960s, a period that ended with the treaty banning nuclear weapon tests for military and for peaceful purposes, in the atmosphere, in outer space, and

[1]Monazite is a phosphate mineral containing rare earth metals like thorium, cerium, lanthanum and neodymium. It is the primary source of thorium, which content can be up to 20–30 % sometimes.

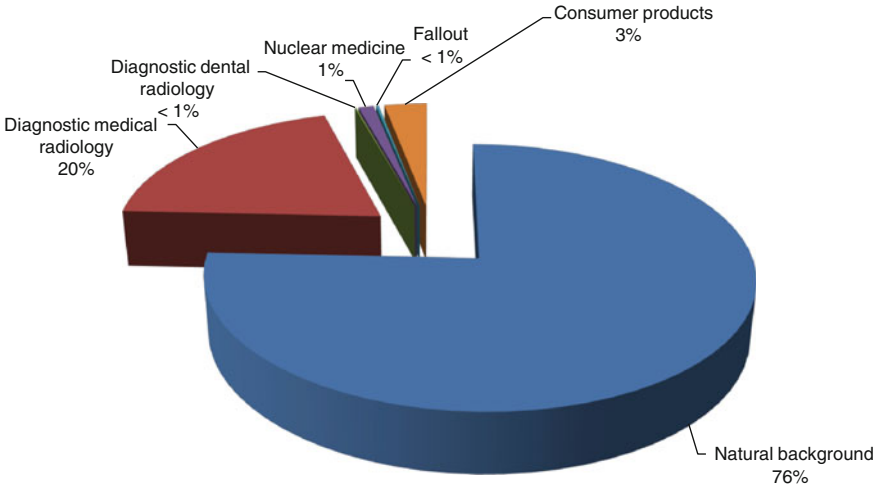


**Fig. 13.3** Average annual per caput dose to the global population

underwater. Important long-lived artificial radionuclides from those explosions, such as $^{3}H$, $^{14}C$, $^{137}Cs$, $^{90}Sr$, $^{239}Pu$, $^{240}Pu$, and $^{241}Am$, were dispersed and deposited all over the earth, including the most remote sites [7]. The annual dose to the world population from the global fallout reached a maximum of 0.14 mSv in 1963, when the Partial Test Ban Treaty was signed, and decreased by almost an order of magnitude by 1979. Currently, the estimated annual per caput effective dose to population due to global fallout is about 0.005 mSv [8].

Space vehicles such as satellites and deep space spacecrafts may also carry nuclear power sources, such as a small nuclear reactor, a radioisotopic thermoelectric generator (RTG), and heating unit (RHU) containing plutonium as options for power and propulsion [9, 10]. Thus, spacecrafts launch accidents and uncontrolled reentries are also potential threats. Some incidents taken from various sources [11–14] and their consequences are illustrated in Table 13.1.

The heat from the friction of the air normally burns up a little satellite as it falls toward Earth at thousands of miles per hour, but bigger objects might not entirely burn up before reaching the ground; though, the primary risk arises from the casualties that can be derived from the direct impact of falling fragments or debris on people, buildings, or vehicles (aircraft ship or train), and the risk of radioactive contamination.

Consequently, international agreements have established safety guidelines, including stringent design, operation, and safety criteria for nuclear power sources, both RTGs and nuclear reactors, used in outer space devices. The Office for Outer Space Affairs (UNOOSA), which implements the decisions of the Committee on the Peaceful Uses of Outer Space (COPUOS), in 1992, established a set of principles applicable to nuclear power sources in outer space which were approved by the General Assembly [15]. These principles were complemented in 2009 by an

**Table 13.1** Examples of launch and reentry space vehicles incidents

| Date | Incident | Place and consequences |
|---|---|---|
| April 1964 | Transit-5BN-3 navigation satellite failed to reach orbit when launched | The spacecraft burned up over Madagascar and the plutonium fuel was injected into the upper atmosphere as it was designed to |
| May 1968 | Nimbus B-1 weather satellite exploded when the launch vehicle had to be intentionally destroyed and the lift off aborted shortly after launch | The remains of the satellite and the RTG plunged into the Pacific Ocean off California; five months later, the RTG and its plutonium dioxide were recovered from the bottom of the Santa Barbara channel |
| January 1978 | Satellite Cosmos 954 reactor core failed to separate and boost the spacecraft to a higher, nuclear-safe orbit as planned | The satellite broke up into hundreds of pieces, many of them quite large. The nuclear reactor reentered over Pacific Ocean and crashed near Great Slave Lake in northern Canada. Major pieces of Cosmos 954 remained intact and impacted the ground, scattering radioactive debris far and wide. The bulk of the $^{235}$U itself survived reentry, contaminating the landscape over a 370 mile long path between Great Slave Lake and Baker Lake in the Northwest Territories, Alberta and Saskatchewan. Pieces scattered across the tundra in a frozen, sparsely populated region |
| July 1979 | Skylab space station burnt-up over the Indian Ocean and Western Australia | Some large chunks survived reentry, making landfall southeast of Perth and elsewhere |
| February 1983 | Cosmos 1402 core reactor failed to separate into high Earth orbit as planned | The satellite reentered the atmosphere and the reactor was the last piece to come Earth. It landed somewhere in the South Atlantic Ocean, and dispersed the radioactive materials over more than 100,000 km$^2$ before sinking to the ocean floor |
| February 2003 | Space shuttle Columbia made an uncontrolled return to Earth at the end of its STS-107 mission | Columbia broke apart over Northeastern Texas, all seven astronauts aboard were killed, and the 100-ton orbiter was destroyed raining debris over Texas and Louisiana |

international safety framework jointly prepared by the Scientific and Technical Subcommittee of the United Nations Committee on the Peaceful Uses of Outer Space and the International Atomic Energy Agency, to reduce the probability of potential accidents that could release radioactive material, and to reduce the magnitude of potential releases and their potential consequences to people and the environment [16].

The two more important radionuclides released in the Chernobyl nuclear power plant accident were $^{131}$I (half-life 8 days) and $^{137}$Cs (half-life 30 years); these two radionuclides were responsible for most of the radiation dose incurred by the members of the general population. The activities released were ~1,760 and 85 PBq respectively (1 PBq = $10^{15}$ Bq). Other releases were 6,500 PBq of $^{133}$Xe, with a half-life 5.25 days, and about 10 PBq of $^{90}$Sr, with a half-life of 28.8 years. Consistent with the fuel particle release within about 20 km of the Chernobyl nuclear power plant, radionuclides like $^{95}$Zr, $^{99}$Mo, $^{144}$Ce, $^{239}$Np, $^{241}$Pu, etc., were deposited in the vicinity of the damaged reactor, although, in small amounts. The average effective dose received by the evacuees (115,000 people) was 31 mSv, while the average effective dose to the inhabitants of contaminated areas of Belarus, Russia, and Ukraine (6,400,000 people)—where $^{137}$Cs levels on soil were greater than 37 kBq/m$^2$—was 9 mSv. The average effective dose to the inhabitants of European distant countries as a consequence of the accident was 0.3 mSv [17].

In the case of the Fukushima Daiichi nuclear power plant accident, noble gases were a significant part of the early releases; it is estimated that around 6,000–12,000 PBq of $^{133}$Xe were released. The mean total activity of $^{131}$I released was around 100–400 PBq, and that of $^{137}$Cs was around 7–20 PBq. These releases are estimated to be approximately one tenth of those from the accident at the Chernobyl nuclear power plant. In this case, most of the releases were dispersed over the North Pacific Ocean and fell on the oceanic surface layer [18]. The doses to the general public, those both incurred during the first year and estimated for their lifetimes, are generally low or very low, comparable with the range of effective doses incurred due to global levels of natural background radiation [19].

The value: Fallout < 1 %, in Fig. 13.3, accounts for the current fallout from nuclear tests in the atmosphere, together with exposure to global population due to unplanned satellite's reentries, to releases from accidents like Chernobyl and Fukushima, and to releases from nuclear power plants.

During high-altitude commercial flights, the estimated passenger dose due to cosmic radiation is low (0.3–60 μSv per flight, depending on latitude and duration) [2], but people who fly frequently probably add some contribution to their overall exposure. Frequent flyers could be exposed to close to 1 mSv in addition to the dose they receive from natural background. If you calculate the dose using the Federal Aviation Administration computer program CARI-6 [20] for a flight between Hong Kong and New York, it may result in doses about 0.1 mSv, depending on the solar activity. Between Hong Kong and Vancouver, the dose may be nearly 0.05 mSv.

Activities associated with ore extraction and processing, other than those associated with the extraction of uranium, can cause enhanced levels of naturally occurring radioactive material (NORM[2]) in products, byproducts, and waste, that

[2]NORM is used more specifically for all naturally occurring radioactive materials where human activities have increased the potential for exposure compared with the unaltered situation.

may affect public exposure [21]. NORM may be present in coal mines and power generation from coal; in metal mining and smelting; the oil and gas industry; the fertilizer (phosphate) industry; rare earth and titanium oxide industries; in tile and refractory industries; and in applications using natural radionuclides like radium and thorium, etc.

The general public can also be exposed through the agricultural use of sludge from water treatment plants [22] or the use of NORM residues [23], as a component of either landfill material or construction material. The typical annual effective dose to workers from all industries is 1 mSv, with the exception of the industry sector dealing with monazite and extraction of rare earths. Doses received by members of the public range from few μSv to fractions of mSv [21, 24].

Consumer products may also contribute to public exposure. Many daily used products contain low levels of radionuclides, added deliberately because of their chemical and/or radioactive properties [2]. These include radio luminous wristwatches containing $^{3}H$ and $^{147}Pm$; other self-luminous products; vacuum tubes; smoke detectors using $^{241}Am$; compact fluorescent light bulbs containing $^{147}Pm$; uranium glazed wall tiles and ceramics; piezoelectric ceramics; incandescent gas mantles; geological specimens; glassware and camera lenses containing uranium, thorium and/or $^{40}K$; thoriated welding rods; antistatic devices using $^{241}Am$ or $^{210}Po$; tritium signs, etc. The annual effective doses during the normal life of these products in the United States range from less than 0.01–10 mSv [2]. The annual dose distribution in Fig. 13.3 shows the relative contribution of consumer products to public exposure.

Medical exposure is accepted to bring more benefits than risks; yet, over the past two decades (1997–2007), the growing trend in diagnostic radiology observed in developed countries due to the high level of innovation and the introduction of new imaging techniques, including computed tomography (CT), angiographic examinations, interventional radiology, and cardiovascular studies, has led to an increase of public exposure. The rapid adoption experienced in photostimulable phosphor imaging is also leading to an increase in the frequency of radiological examinations. As shown in Fig. 13.3, medical exposures now contribute around 20 % to the average annual per caput dose to the global population [25].

The above accounted facts are a good reason to require yearly constraints from each controllable man-made source, other than medical, to not further contribute to public exposure and observe the limit of 1 mSv to the representative person[3] in one single year.

[3]A representative person is an individual receiving a dose that is representative of the more highly exposed individuals in the population.

## 13.3 Public Exposure Control

Constraints to planned exposures situations, along with other optimization measures, such as the appropriate shielding, containment, and confinement of radiation sources, are, in general, sufficient to control public exposures as well. However, to protect the general public from the potential of radiation exposures, the radiation protection program should also include specific actions: (1) to properly manage and storage spent and/or disused sealed sources, and radioactive waste; (2) to meet the requirements for the safe transport of radioactive materials; and, (3) to schedule and perform a proper surveillance over the liquid and gaseous effluents to be released to the environment, if applicable.

In an existing exposure situation—e.g., a contaminated site from a radiological accident, a residual radioactive material from a past radiological event, radon in dwellings, etc.—the measures to control public exposures are aimed at reducing the existing hazards based on a previous safety assessment; at establishing restrictions for the use or access, e.g., food and water consumption, land use, exclusion of people from an area, etc.; and/or at dealing with the waste from remedial works, e.g., treatment, recovery, recycle, and disposal.

Regarding NORM and consumer products, efforts should be made to identify industry sectors and process materials most likely to require regulations to reduce the effective dose to population. Recycling or recovery may be a way to diminish the quantities of NORM waste to be disposed of. On the subject of radon exposure and building materials, there are technical requirements in the form of building codes and construction practices to limit the entry of radon into building dwellings and restrict the use of specific building materials.

Concerning emergency exposure situations, the most important provisions of the radiation protection program to control public exposure are intended for conducting regular safety assessments to reduce the likelihood of failures, or conditions, that could lead to a loss of control over the source, and, in case they occur, to respond to it and minimize its consequences.

In medical applications, the justification principle should prevail taken into account all relevant information, as well as the benefits and risks of alternative techniques available, when using specific diagnostic and interventional radiologic procedures. The radiation protection program should also include specific arrangements and dose constraints to protect visitors, caregivers, and comforters from avoidable exposures by inpatients undergoing treatment with radiopharmaceuticals and/or brachytherapy sources. Dose constraints may also be applied, on a case-by-case basis, to volunteers for biomedical research.

## References


1. U.S. NRC (2014) Sources of radiation. United States Nuclear Regulatory Commission. http://www.nrc.gov/about-nrc/radiation/around-us/sources.html
2. UNSCEAR 2008 (2010) Sources and effects of ionizing radiation. Report to the general assembly with scientific annexes, vol I, Annex B Exposures of the public and workers from various sources of radiation. UN, New York
3. UNSCEAR (1993) Sources and effects of ionizing radiation. Report to the general assembly. Annex A: Exposures from natural sources of radiation. UN, New York
4. Canadian Nuclear Safety Commission (2014) Natural background radiation. Canadian Nuclear Safety Commission. http://nuclearsafety.gc.ca/eng/resources/fact-sheets/natural-background-radiation.cfm
5. UNSCEAR (2000) Sources and effects of ionizing radiation. Report to the general assembly with scientific annexes, Volume I, Annex B, Exposures from natural radiation sources. UN, New York
6. Hendry JH et al (2009) Human exposure to high natural background radiation: what can. J Radiol Prot 29:A29–A42
7. UNSCEAR (2000) Source and effects of ionizing radiation. Report to the general assembly, Volume I, Annex C Exposures from man-made sources of radiation. United Nations, New York
8. UNSCEAR (2008) Sources and effects of ionizing radiation. Report to the General Assembly. Sixty-thrid session. Supplement No. 46. United Nations, New York
9. World Nuclear Association (2015) Nuclear reactors and radioisotopes for space. World Nuclear Association. http://www.world-nuclear.org/info/non-power-nuclear-applications/transport/nuclear-reactors-for-space/
10. IAEA (2005) The role of nuclear power and nuclear propulsion in the peaceful exploration of space. IAEA, Vienna
11. Listverse (2012) Top 10 space age radiation incidents by Patrick Weindinger. Listverse Ltd.. http://listverse.com/2012/01/20/top-10-space-age-radiation-incidents/
12. Space.com (2015) 6 biggest spacecraft to fall uncontrolled from space by mike wall. Purch. http://www.space.com/13049-6-biggest-spacecraft-falls-space.html
13. Erickson LK (2010) Space flight: history, technology, and operations. Government Institutes, Lanham
14. Hagen R (1998) Nuclear powered space missions—past and future. http://www.space4peace.org/ianus/npsm3.htm#kosmos198
15. United Nations Office for Outer Space Affairs (2015) Treaties and principles, resolution adopted by the general assembly 47/68. Principles relevant to the use of nuclear power sources in outer space. UNOOSA. http://www.unoosa.org/oosa/en/ourwork/spacelaw/principles/nps-principles.html
16. COPUOS, IAEA (2009) Safety framework for nuclear power source applications in outer space. IAEA, Vienna
17. UNSCEAR (2008) Sources and effects of ionizing radiation. Report to the general assembly. Volume II Annex D Health effects due to radiation from the Chernobyl accident. United Nations, New York
18. IAEA STI/PUB/1710 (2015) The Fukushima Daiichi accident. Report by the Director General. IAEA, Vienna
19. UNSCEAR 2013 Report (2014) Sources, effects and risk of ionizing radiation. Volume I Scientific Annex A. Levels and effects of radiation exposure due to the nuclear accident after the 2011 great east-Japan earthquake and tsunami. UN, New York
20. Federal Aviation Administration (2015) CARI-6: radiobiology research team. U.S. Department of Transportation, Federal Aviation Administration. https://www.faa.gov/data_research/research/med_humanfacs/aeromedical/radiobiology/cari6/

21. IAEA STI/PUB/1326 (2008) Naturally occurring radioactive material (NORM V). In: Proceedings of the fifth international symposium on naturally occurring radioactive material, 19–22 March 2007, Sevilla
22. Kleinschmidt R (2011) Radioactive Residues Associated with Water Treatment Use and Disposal in Australia. Queensland University of Technology, Brisbane
23. Committee on Evaluation of EPA Guidelines for Exposure to Naturally Occurring Radioactive Materials; National Research Council (1999) Evaluation of guidelines for exposures to technologically enhanced naturally occurring radioactive materials. National Academies Press, Washington
24. International Atomic Energy Agency (2005) Naturally occurring radioactive materials (NORM IV) IAEA-TECDOC-1472. In: Proceedings of an international conference held in Szczyrk, Poland, 17–21 May 2004, Szczyrk
25. UNSCEAR (2008) Sources and effects of ionizing radiation. Report to the general assembly with scientific annexes, Volume I, Scientific Annex A medical radiation exposures. UN, New York

# Chapter 14
# Radioactive Waste Management

Radioactive waste is any material—liquid, solid or gas—without further use, which contains or is contaminated with radionuclides at concentrations or activities exceeding the clearance levels established by the regulatory authority [1].

But any material containing or contaminated with radionuclides is radioactive regardless its activity or radionuclide concentration from a physical point of view. That is why to define a radioactive waste, and for regulatory purposes, it is required to emphasize the boundary below which any further regulatory control is impracticable and, a material already under regulatory control can be removed from this control in the form of a waste. This boundary is the clearance level [2].

## 14.1 Clearance Levels

It was learned that the term exemption is used to establish if a radiation source, by its nature—whether naturally occurred or man-made—should or should not be under regulatory control. Exempted quantities, meaning by them the individual quantities of byproduct materials that can be received, possessed, used, transferred, owned, or acquired without the requirement of a license, are set forth in 10 CFR § 30.71 Schedule B [3, 4]. Automatic exemption levels recommended by the IAEA are listed in tables I.1 and I.2 of the International Basic Safety Standards [5]. Once a source is declared exempted, a notification or authorization is not needed to have or make use of it (See Chap. 9).

The term clearance was defined on the other hand to establish which radioactive material, already under regulatory control, can be removed from this control [2, 5].

The importance of clearance was brought into attention while determining the subsequent reuse, recycle or landfill disposal of some residual materials—e.g., metals, soil, building rubble, equipment, etc.—containing or contaminated with radionuclides, produced by nuclear plant decommissioning or remedial actions like decontamination or restoration after a radiological or nuclear accident.



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Clearance may be granted for specific situations taking into account the physical or chemical form of the material and its use, or the means of its disposal, that is, the triviality of the failure risk at the time and location of the release. Clearance levels are hence established in terms of low enough activity concentration per unit mass, or per unit surface area, to allow such material to be used or disposed of without any further restriction.

Consistent with the International Basic Safety Standards [5], a material can be cleared without further consideration if, in all reasonably foreseeable situations, the effective dose expected to be incurred by any individual due to the cleared material is of the order of 10 μSv or less in a year, or 1 mSv, in case of low probability scenarios. The activity concentrations of radionuclides of artificial origin recommended by the IAEA for clearance of solid material are listed in Table I.2 of the Basic Safety Standards.

A calculation of potential annual doses to an individual following the clearance of scrap iron and steel, copper, aluminum, and concrete rubble from licensed nuclear facilities in the United States can be found in the Report NUREG 1640 [6]. These results are expressed in terms of effective dose from 1 year of exposure per unit activity in a gram, or on a square centimeter, of cleared material for each separate radionuclide in each material. In this Report, a total of 86 exposure scenarios were assessed for 115 radionuclides considered potential components of the residual radioactivity in these materials. The models and scenarios can be adapted to specifically fit the situation at hand.

## 14.2 Radioactive Waste Management

Radioactive waste management includes all administrative and operational activities required for the handling, pretreatment, treatment, conditioning, transport, storage and disposal of radioactive waste [7]; all aimed to protect the public and the environment from avoidable exposures. The main steps of radioactive waste management are shown in Fig. 14.1.

Pretreatment is the initial step after waste generation. It commonly consists of collection, segregation, chemical adjustment and decontamination of reusable materials, and can also include a storage period for decay. In this stage, waste is segregated into streams to be handled in a similar way, e.g., compactable and combustible solids, aqueous and organic or toxic liquids, etc. To minimize its volume, radioactive waste should also be collected apart from exempt or nonradioactive residual materials, and from waste that can meet the clearance level for reuse or discharge.

There are three general principles implied in the managing of radioactive waste: dilute and disperse; concentrate and contain; and delay and decay [8].

Dilute and disperse are commonly applied to gaseous or liquid streams when clearance levels can directly be met by dilution or dispersion. For example, 10 CFR part 20 [3] authorizes the release of liquids into the sanitary sewerage if the total

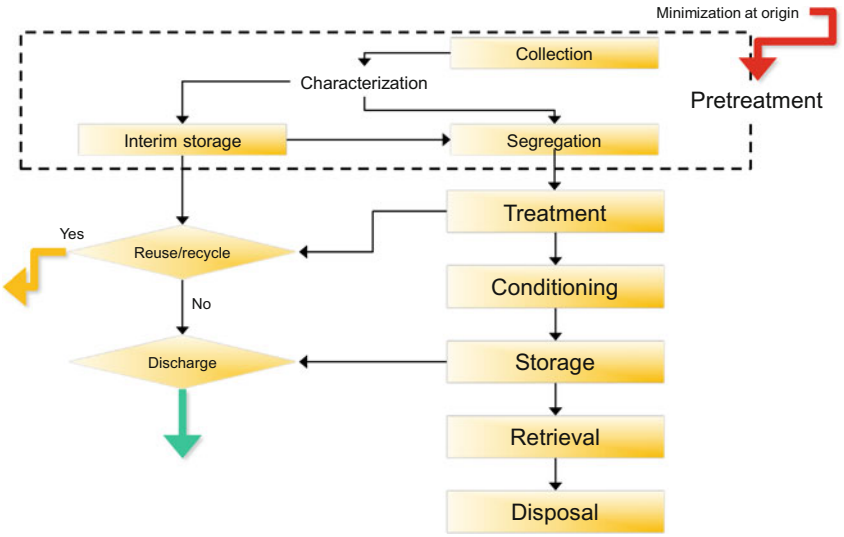


**Fig. 14.1** The basic stages of radioactive waste management

radioactivity released in a year does not exceed 185 GBq of $^{3}H$, 37 GBq of $^{14}C$, and 37 GBq of all other radionuclides combined. Most gaseous effluents reach acceptable levels while passing through high-efficiency filter systems and can as well be diluted and dispersed into the atmosphere. 10 CFR part 20 also provides concentration limits for airborne effluents released to the environment. These values are equivalent to the concentrations which, if inhaled continuously over the course of a year, would produce a total effective dose of 50 μSv.

Delay and decay are usually applied to waste containing short-lived radionuclides; they are then stored and isolated until decaying into accepted harmless levels for reuse, recycle, or discharge. A decay-in-storage period of ten half-lives will reduce the activity of the waste by a factor of approximately 1000. Some longer lived waste with higher activity levels can also be stored for cooling before treatment.

Concentrate and contain involve operations to change the waste amount and composition by volume reduction, radionuclide removal, or chemical reaction; and its further confinement by conditioning and packaging. Typical treatment processes include compaction of compressible dry solid waste, and incineration of combustible solid waste or organic liquid wastes for volume reduction; evaporation to reduce the volume of aqueous liquid waste; liquid waste filtration, or ion exchange for radionuclide removal; and precipitation or flocculation of chemical species to change their chemical form. Low-level effluents resulting from treatment can also be cleared for recycle or reuse.

Conditioning involves operations to transform radioactive waste into a stable solid form that is insoluble, prevents dispersion of radionuclides in the environment, and is suitable for handling, transportation, storage, and disposal. Conversion of radioactive waste into a stable solid form can be made by immobilization in

cement or through vitrification in a glass matrix. Immobilized waste can then be placed in a steel drum or other engineered container to create a waste package that meets all the requirements.

Conditioned waste might be stored for many years before it undergoes further processing and final disposal; hence the need of interim storage facilities designed to isolate the waste for a limited period of time, protect the public and the environment, and control the waste decay, retrieval and disposal. Retrieval involves recovering waste packages from storage either for inspection, disposal, or further storage in new facilities.

Disposal implies the emplacement of radioactive waste into an appropriate facility or location without the intention of retrieval [1]. Long-term storage, which can be of hundreds or thousands of years, requires to achieve safety by containment and isolation of properly conditioned wastes using multiple barriers—natural or engineered—to ensure that no release of radionuclides to the environment will occur and the overall activity will decay during the estimated period of time.

Radioactive waste management safety objectives are hence aimed first at keeping radioactive waste generation to the minimum practicable, and second, at providing an acceptable level of protection to human health and the environment in all operations, now and in the future. The predicted impacts on the health of future generations should be no greater than the relevant levels of impact that are acceptable today [9].

## 14.3 Classification of Radioactive Waste

Radioactive waste is generated in nuclear energy and fuel cycle industries, academic, medical, research and industrial uses of radiation sources, military industry, and in activities associated with mineral ores that contain naturally occurring radionuclides—e.g., phosphate ores, oil, or gas drilling. Different classification criteria are therefore followed to segregate radioactive waste. Origin, physical form, activity level, half-life, radionuclide concentration, chemical form, radiation type, and/or a specific regulatory, operational and disposal requirement might be the criterion to characterize and classify the waste.

Waste streams considerably vary in composition and properties depending on their origin. Radioactive waste is produced from the mining, milling, conversion, and enrichment of uranium, fuel production and fuel reprocessing, to nuclear power plant operation, nuclear research and development, radioisotope production, processing of mineral ores or other materials containing naturally occurring radionuclides, and different applications in industry, agriculture, research, education, and medicine.

At the generation point, the logical and simplest distinction between waste streams should be first its physical form, i.e., solid, liquid, and gaseous.

Solid waste is wide ranging from equipment, tools, and structures radioactively contaminated to used filters, glassware, gloves, aprons, masks, paper towels,

syringes, needles, plastic sheets and bags, etc.; in addition to animal corpses; excretes; and several similar objects which can be generated in different industrial or field settings, hospitals, clinics, laboratories, research and academic institutions, etc. Yet it is also practical to further classify solid waste as:

- Dry solid heterogeneous waste, waste which is compactable or combustible;
- Hard materials, items whose size normally calls for shredding;
- Biological waste, animal carcasses, excretes, organ and tissues, and any other material able to rot;
- Sharp objects, e.g., wires, needles, scalpel and razor blades, blood lancets, etc.; and
- Mixed waste, also flammable, corrosive, toxic or reactive.

Spent and disused sealed sources—i.e., those sources that are no longer in use or have decayed to an activity that is unfit for the specific purpose—are a very important type of solid waste that requires special consideration [10]. Due to the concentrated nature of their radioactive content, these sources must be safely stored, isolated, and disposed of. Spent and disused sealed sources can also contain radionuclides like $^{137}$Cs, $^{60}$Co, $^{192}$Ir, $^{90}$Sr, $^{226}$Ra, $^{241}$Am, $^{238}$Pu, $^{239}$Pu, etc., in relative high activities, not suitable for near surface disposal.

Wet solids—also called wet waste—are another type of solid waste which includes spent ion exchange resins, filter media, sludge, concentrates, sediments, etc., arising from aqueous liquid treatment systems [11].

Liquid waste could vary from small volumes of, e.g., scintillation counting samples, discarded radiopharmaceuticals, surplus solutions, contaminated solvents, blood or body fluids, to large volumes from, e.g., coolant, cleanup, and sanitary systems. Based on the origin and chemical composition, liquid waste could be classified in two broad categories: (1) aqueous radioactive liquid in which the waste materials are either dissolved or evenly distributed in water; and (2) mixed radioactive liquid containing other hazardous chemicals, such as organic flammable or toxic solvents, heavy metals, corrosive chemicals, substances that pose biological hazard, etc.

Liquid waste from academic and biomedical research laboratories typically contains $^{14}$C, $^{3}$H, $^{125}$I, $^{57}$Co, $^{32}$P, $^{45}$Ca, $^{35}$S, and $^{75}$Se. The use of radiopharmaceuticals in medical diagnostic usually generate liquid waste containing $^{99m}$Tc, $^{111}$In, $^{113}$Sn, $^{123}$I, $^{67}$Ga, $^{75}$Se, $^{133}$Xe, $^{201}$Tl, and $^{203}$Hg, among other radionuclides. $^{131}$I, $^{32}$P, $^{89}$Sr, and $^{90}$Y are mostly present in liquid waste from medical therapy, while long-lived radionuclides—among them $^{137}$Cs, $^{60}$Co, $^{54}$Mn, $^{59}$Fe, $^{90}$Sr, and $^{110m}$Ag—are present in liquid waste from NPP.

Gaseous waste is in the form of aerosols and volatile elements—halogens, noble gases, tritium, and $^{14}$C. Different filters, absorbers and electrostatic precipitators are used to clean the airstreams from contaminants and meet the discharge limits. As a result, a secondary waste is produced in solid or liquid form [12]. High-Efficiency Particulate Air (HEPA) filters and/or absorber filters, e.g., activated charcoal for iodine, are usually used at the exit of fume hoods and glove box extract systems,

chambers and other enclosed containments, as well as supply and extract ventilation systems in nuclear plants, laboratories, hospitals, irradiation chambers, etc.

Regarding radionuclide half-life, solid, liquid, and gaseous waste is classified as long-lived waste ($T½ > 30$ years) and short-lived waste ($T½ < 30$ years). Since radiological hazard of short-lived wastes can be reduced over a few hundred years to acceptable levels by radioactive decay, it is very important to segregate the waste by half-life as well. An additional category of very short-lived waste for decay-in-storage is also introduced to differentiate waste containing only radionuclides with half-lives less or equal than 100 days [13]. As stated by § 35.92 10 CFR Part 35 [14], the half-life for decay-in-storage should be $\leq 120$ days. After decay, very short-lived waste can be discharged as ordinary trash or medical waste, as appropriate. $^{132}$Ir (73.83 days), $^{131}$I (8.02 days), $^{201}$Tl (3.04 days), and $^{99m}$Tc (6 h) are examples of very short-lived radionuclides.

The classification of waste by radiation level or activity level is designed for final disposal. In correspondence to it, waste is classified in Very Low-Level Waste (VLLW), Low-Level Waste (LLW), Intermediate-Level Waste (ILW), and High-Level Waste (HLW) [13].

Very Low-Level Waste (VLLW) is a waste with limited hazard and an activity level of just one or two orders above the exemption level. This type of waste does not need the most stringent requirements of containment and isolation and, therefore, is suitable for disposal in near surface landfill-type facilities with limited regulatory control. VLLW mostly arises from the decommissioning of nuclear power plants, from mining or processing of ores and minerals, and from conventional industry—oil–gas exploration, phosphates, etc.

Low-Level Waste (LLW) is a waste with limited amounts of long-lived radionuclides. The allowable average activity concentration of long-lived beta and/or gamma emitting radionuclides, such as $^{14}$C, $^{36}$Cl, $^{63}$Ni, $^{93}$Zr, $^{94}$Nb, $^{99}$Tc and $^{129}$I, can be up to tens of kBq per gram. Almost all the waste produced in medical, academic, research and industrial applications, including reactor operation, and power plant decommissioning is LLW; hence, represents the larger volume and most variable composition. LLW is suitable for disposal in engineered near surface facilities; existing facilities are built at varying depths, from surface down to 30 m, and requires containment and isolation for hundreds of years.

Intermediate-Level Waste (ILW) is a waste that contains long-lived radionuclides in quantities that require a superior degree of containment and isolation than that provided by near-surface disposal. A depth of between few tens and few hundreds of meters is indicated for ILW if both natural barriers and engineered barriers are selected properly. The likelihood of human intrusion is considered to be significantly reduced at such depths.

High-Level Waste (HLW) is a waste that generates significant amounts of heat, contains large concentrations of both short and long-lived radionuclides, and requires the highest degree of containment and isolation to ensure waste long-term safety. Such containment and isolation is generally understood that could be provided by disposal in deep stable geological formations, several hundred meters or

more below the surface, with engineered barriers. Heat dissipation has to be taken into account in the design of the disposal facility.

Regulations that control radioactive waste in the United States are based primarily on the type of enterprise that produced it, i.e., the origin of the waste, rather than waste's actual radiological characteristics and properties. Roughly, the Department of Energy (DOE) controls the radioactive waste related to nuclear weapons production and certain research activities, while the NRC and agreement states regulate commercial radioactive waste resulting from the production of electricity, the use of radionuclides in industry, medicine and research, and the processing of mineral ores or other materials containing naturally occurring radionuclides [15]. EPA has jurisdiction on certain aspects of the disposal of radioactive waste as well.

The NRC distinguishes three basic types of commercial radioactive waste: high-level waste, mill tailings, and low-level waste. High-level radioactive waste basically consists of spent nuclear reactor fuel; mill tailings are the residues remaining after the processing of natural ore to extract uranium and thorium; and low-level waste is any commercial radioactive waste that is not a high-level waste or a uranium and thorium milling waste [16].

High-level radioactive waste and low-level radioactive waste are subject to the requirements specified in 10 CFR Part 60 [17] and Part 61 [18] respectively. By origin, high-level radioactive waste (HLW) includes spent nuclear fuel (SNF), high-level waste other than spent fuel from past fuel reprocessing activities (HLW), and transuranic radioactive waste (TRU). Low-level radioactive waste (LLW) is, in its turn, defined by what is not, i.e., radioactive material that is not HLW, SNF or byproduct [19, 20]. Byproduct material is any radioactive material—except enriched uranium or plutonium—produced by a nuclear reactor, the tailings or wastes produced by the extraction or concentration of uranium or thorium or the fabrication of fuel for nuclear reactors, any material that has been made radioactive through the use of a particle accelerator, any discrete source of $^{226}$Ra, and any source of NORM, other than source material [21].

LLW is further divided based on disposal options into two broad categories: waste that qualifies for near-surface burial and waste that requires deeper disposal. As summarized in Table 14.1, NRC classifies commercial LLW that qualifies for near-surface disposal as Class A, B, or C. Key decision parameters in this classification system are the waste form and packaging physical stability, and its radionuclide concentration [18]. Waste that requires deeper disposal is known as Greater-Than-Class C (GTCC) LLW. It includes activated metals and sealed spent and disused sources.

LLW involves large volumes of waste produced by a variety of different processes, including the nuclear fuel cycle, medical and/or biotechnological research, medical, industrial, academic and agricultural applications, the production of radioactive chemicals and drugs, the manufacture of commercial products, as well as government military operations. Accordingly, it is possible that at commercial disposal facilities and, as appropriate; LLW is further classified into various waste

**Table 14.1** Radioactive waste classification for near-surface disposal

| Class | Level of radiation | Packaging stability | Examples | Intruder protection |
|---|---|---|---|---|
| A | Low, no shielding required | Separate disposal cell | Trash, soil, rubble, depleted uranium, mildly contaminated equipment and clothing | Waste decays to acceptable levels to intruder after 100 y |
| B | 10–40 times greater than Class A, shielding required | 300-y stabilization requirement | Reactor components, sealed radioactive sources, filters and resins from nuclear power plants | Waste decays to acceptable levels to intruder after 100 y, provided that waste form is recognizable |
| C | 10–100 times greater than Class B, shielding required | 300-y stabilization requirement | Same as Class B, but with higher activity | Requires deeper disposal (or barriers) to protect intruder, waste decays to acceptable levels to intruder after 500 y |
| Greater than C | Greater than Class C | Unspecified by regulations | Reactor components and filter resins from reactor decommissioning, sealed sources | Generally not acceptable for near-surface disposal, geologic repository disposal |

generator categories, e.g., academic, commercial (for-profit entity other than health, utility, or academic), government, health, and utility.

A recent report of the National Academies of Sciences, aimed at improving the regulation and management of low activity radioactive waste, defines low activity waste as any waste containing radioactive materials that fall well within the NRC classification system for low-level waste; slightly contaminated solid materials—debris, rubble, and contaminated soils from nuclear facility decommissioning and site cleanup—; discrete sources; uranium and thorium ore processing waste; naturally occurring radioactive material waste arising from the recovery of natural resources—extraction of rare earth minerals and other mining operations, oil, and gas—and water treatment, as well as defense low-level waste [15].

According to the Manifest Information Management System (MIMS), a database used to monitor the management of commercial LLW in the U.S., the total volume of waste shipped in 2014 to currently operating commercial LLW disposal facilities was 32,122 $m^3$, 98 % of them Class A [22]. The same data for the period 2005–2014 is shown in Fig. 14.2.

A decreasing trend in yearly waste volume is clearly noticed in Fig. 14.2; fluctuations may be largely influenced by one-time-only events, for example, decommissioning projects, cleanup activities, nuclear power plant outages, and source manufacturing projects, etc.

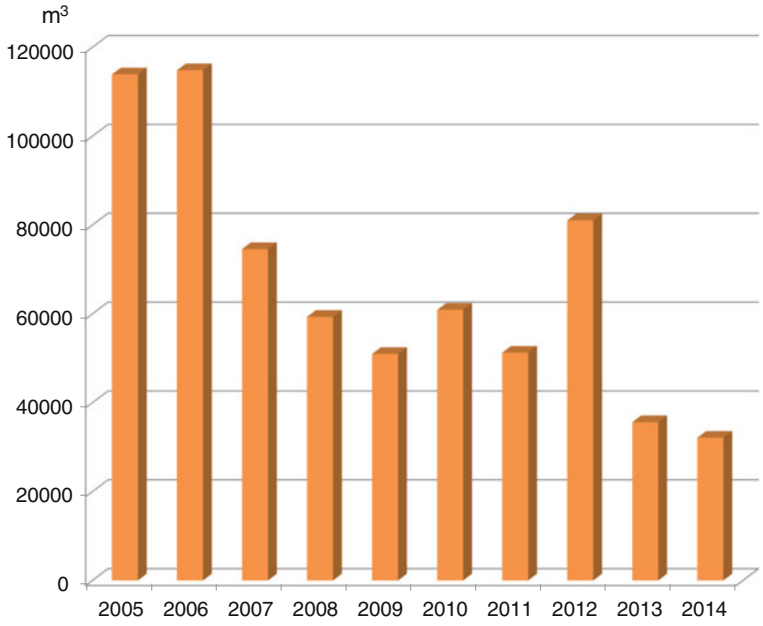


**Fig. 14.2** Low-level radioactive waste received by operating commercial LLW disposal sites

## 14.4 Pretreatment Procedures

To lessen the waste volume at the generation point, the first step is to assess in advance the minimum amount of waste that can be produced, and design operation procedures that prevent contamination. A key factor is to implement best practices for the receipt, inventory, storage, and handling of radioactive material.

Appropriate working conditions are also required to reduce unnecessary waste, e.g., ventilation and filtering systems; floor, wall, ceiling, doors, windows, and benches covered with nonabsorbent, chemical resistant and easily cleaned materials; a clear distinction of clean and unclean tools, instruments, laboratory glassware and equipment; an accurate area classification and signaling, etc.

Written procedures should be anticipated for waste classification, collection, segregation, chemical adjustment, and packaging. Separated liquid collection systems may be requested, e.g., sewer systems, sinks, etc. Solid radioactive waste containers are kept apart from common waste. Sharps that can penetrate the skin—e.g., broken capillary tubes and glass pipettes, blades, glass microscope slides and cover plates, hypodermic and non-hypodermic needles, etc.—are also collected apart in sturdy approved containers.

Radioactive waste is also segregated and labeled by content: radionuclide, half-life, chemical composition, etc. Infected radioactive waste, as well as mixed waste posing radioactive and flammable, corrosive, toxic, explosive, or reactive

hazard—liquid scintillation cocktails, organic solvents, toxic metals, etc.—are also collected and managed apart. Infected waste is usually disinfected prior to storage or treatment. An example of a pretreatment flow chart is show in Fig. 14.3.

Disinfection and decontamination should be taken into account, if appropriate. Decontamination is used to recuperate recoverable equipment and thus reduce the volume of waste that must be disposed of, or to eliminate removable surface contamination when its degree and/or nature can be an obstacle for further

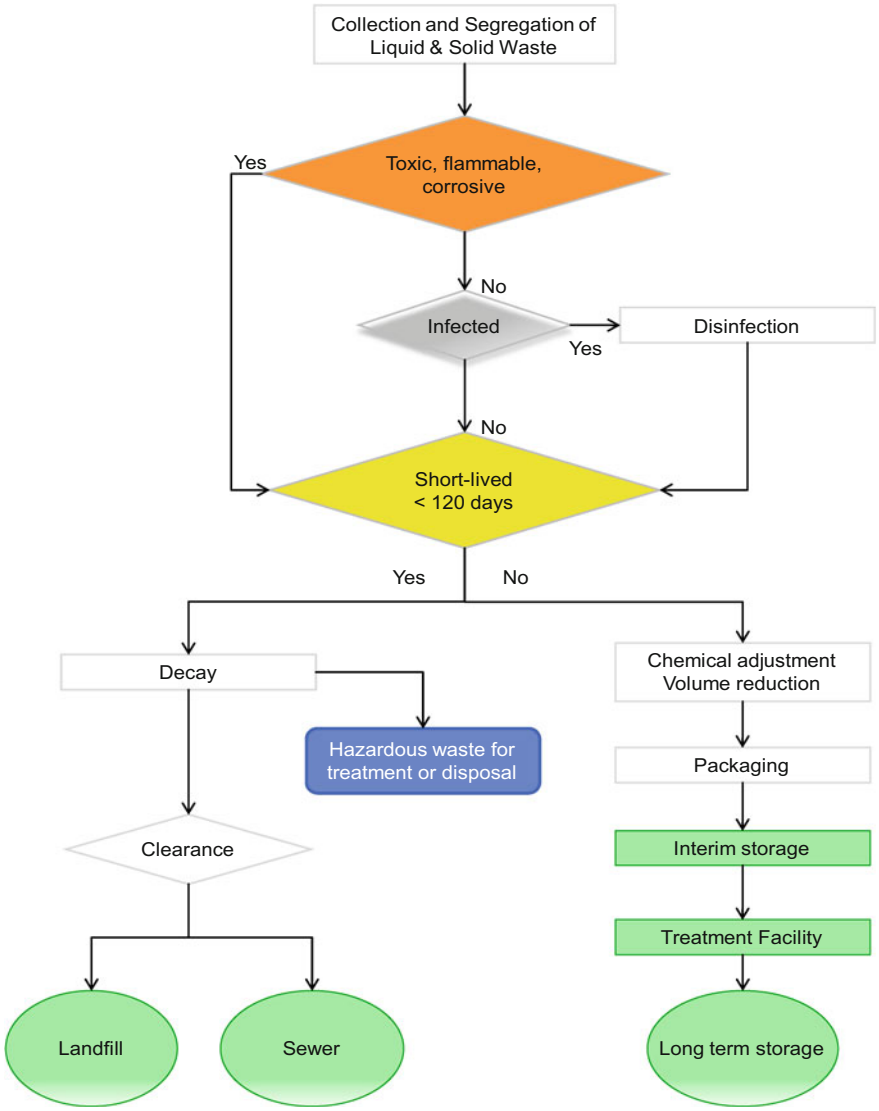


**Fig. 14.3** Pretreatment flow chart

treatment. It is usually performed by washing or using chemical or mechanical techniques. Disinfection is aimed at destroying infectious microorganisms by heat, chemical means, or microwave irradiation. Highly infectious waste, such as cultures and stocks of infectious agents from laboratory work, should be sterilized by autoclaving at the earliest stage possible.

Waste containing very short-lived radionuclides (less than 100–120 days) is isolated and stored for decay in suitable locations until it can be cleared and released as conventional waste. The following conditions should be assured:

- Procedures to handle and isolate waste for decay-in-storage and subsequent release;
- Sufficient room to handle, classify and accommodate the waste for as long as it decays (3–4 years);
- Adequate shielding to comply with dose requirements;
- Periodic waste sampling and measurement;
- Dose rate control of the site and its surroundings;
- Records of stored wastes, measurements and releases.

After decay, hazardous waste must be managed and handled in compliance with specific regulations, e.g., RCRA Subtitle C in the United States [23].

Chemical adjustment of acidic or alkaline liquids, filtration of suspended materials, and/or volume reduction, might be necessary to prepare the waste before its packaging for temporary storage or transport to a treatment facility. Some solid waste, as paper, plastic, cloth, cardboard, wood, and metals, can be cut or shredded into ribbon-like pieces used to reduce the physical size of a particular waste going to incineration or compaction; while brittle materials, such as glass or concrete blocks, can be crushed into smaller fragments.

Dry solid wastes are collected in yellow colored plastic bags placed in covered containers, trash bins or drums, labeled with the radiation symbol and tagged with necessary data to identify the waste. Refuse cans with foot operated lids are particularly convenient. Collected bagged waste is next packaged in sturdy fiber drums or cardboard boxes lined with plastic, sealed and labeled for temporary storage, further treatment or transport. To collect sharps waste (glass, metal, or plastic with rigid corners, sharp edges, or protruding pieces that can slice, scrape, or pierce the skin), rigid, leak-proof, puncture-resistant sharps containers, lined with plastic, are used. Sharps containers are also packaged in cardboard boxes lined with plastic or shielded boxes, as appropriate. Solid radioactive waste also contaminated with chemicals or infectious agents are labeled and segregated accordingly.

Empty lead pigs and scintillation vials are collected apart in plastic lined containers, sturdy drums or boxes specifically identified for pigs, glass and plastic vials. The container capacity should be enough to collect wastes generated, e.g., daily or weekly—depending on the collection schedule—and should be no more than three quarters filled before sealing. The appropriate shielding is recommended where applicable.

It is convenient to segregate dry solid waste according to their half-life in: (a) 10 h or less; (b) less than 10 days; and less than 100 days, for decay-in-storage and discharge [11]. Solid waste greater than 100 days is further classified in two general groups: (a) compactable and incinerable; and (b) non-compactable and non-incinerable.

Most wet solid waste—filter cartridges, precipitation sludge, charcoal media or filtration media, spent resins, etc.—arise from the cleanup of gaseous and liquid radioactive streams prior to discharge or recycle. This type of waste can lead to undesirable chemical and/or biological reactions while in storage or transit, thus, it should be drained, dewatered or dried to the most thorough extent possible, before place it in any receptacle. There are technical specifications and limitations to the containers used for wet solid waste.

Liquid waste can be discharged directly down the drain only if the material is readily soluble in water, and its radionuclide concentration is at or below the established sewerable limits stated by § 20.2003 10 CFR Part 20 [3]. A good practice is to single out and label one sink for this purpose; leave water running for several minutes after discharge; and survey the area after completion. Records of the release should also be kept. Otherwise, liquid waste, both aqueous and organic, is separately collected in a plastic or chemically compatible covered jug or carboy, labeled with the radiation symbol, placed inside a secondary containment, for example, a bucket or tub, to prevent spills or leaks. Buckets or tubs should be large enough to hold all the liquid in the primary container plus 10 %. Filling the pail with an absorbent for the liquid waste—e.g., diatomaceous earth—could be a requirement as well.

It is a good practice to have one container—jug, carboy—per radionuclide whenever possible and practicable; but if the waste contains two or more radionuclides with different half-lives, the waste should be labeled for the longest-lived. For instance, a mix of $^{35}S$, $^{3}H$ and $^{14}C$ is labeled as $^{14}C$. The appropriate shielding is recommended when applicable. Organic waste, scintillation cocktails, and lubricating oils are each collected in individual containers clearly identified. Suitable records should be maintained to identify the type of waste—aqueous, organic solvent, scintillation liquid, mixed waste, biological, etc.—radionuclide, and activity level in each container.

Larger liquid waste volumes containing chelating and complexing agents arise from cleaning areas—showers, laundry, decontamination—special sewerage system drainages; leakage collectors; cooling systems, etc., e.g., in radioisotope production facilities, nuclear research laboratories, and nuclear reactors. Pretreatment of such waste includes oxidation and reduction, pH adjustment, and steam distillation processes. They may be performed to oxidize organic contaminants, decompose complex substances, alter the valence of elements, or adjust the ionic species. Some waste streams may require coarse filtering and oil/solvent removal.

Animal carcasses, organs, tissues and bedding are double bagged, properly labeled and frozen for further treatment and/or disposal when they are not to be incinerated in situ. Bench blankets for extra wrapping and/or bags filled with sorbent materials, such as sawdust or peat moss, are recommended to avoid moisture

seeping through the bag while the waste is in storage or transit. Plastic bags for biological radioactive waste should be red colored, so as to easily recognize that the radioactive waste poses biological hazard.

## 14.5 Treatment Processes

Treatment is necessary to reduce the volume of waste going to interim storage or disposal, and to make it safer stabilizing radionuclides in less leachable and more durable final waste forms for transport, storage, later retrieval or final disposal. Treatment procedures differ with the type of waste, activity, and half-life; also, with the disposal options locally available.

Combustible dry solid waste, organic solid waste, organic liquids—oils, solvents, scintillation liquids, vials, etc.—and, to some extent, spent resins can be reduced to ashes by incineration; the resulting radioactive concentrated ashes can then be compacted, melted with or without glass, or embedded in a resin cement matrix to further reduce its volume and provide for safe transport and long-term storage [24]. Aside from achieving the highest volume reduction factor—between 80 to 120 times depending on the density of the waste—incineration can also reduce the total chemical toxicity of the waste [25].

The volume reduction factor (VRF) is the ratio of the original waste volume to final waste volume after treatment. Commercial waste treatment facilities are equipped with high technology incineration processes, designed to completely and efficiently burn the waste while producing minimum emissions. Secondary waste—ash, slag, and filter dust—is stored in metal drums.

Hospitals, clinics and research centers may have small controlled incinerators, appropriate for their disinfection needs, which could also be authorized to reduce the volume of certain low-level solid radioactive waste, providing it does not contain radioiodine or any other volatile radionuclide.

Compaction is a reliable volume reduction technology, used for processing a wide range of solid combustible or noncombustible low-level waste—e.g., bags of rubbish, scrap metal, rubble, paper, wood, plastic, elastic materials in small amounts, ash and slag from incineration, glass, metallic parts and pipes, air filters, soil/sand. By this method, the waste is compressed to the ultimately density, i.e., all the void space in the waste is compressed out.

The volume reduction factor (VRF) typically varies from 3 to 10 depending on the nature of the waste and the pressure applied, although some modern technologies can obtain better results. Compactors can range from low-force small compactors of $\sim$50 kN compaction force to super compactors with over 10 MN compaction force. Compactors are built in different designs to match drum dimensions and to avoid environmental contamination. Soft, easily compressible bags containing low-level waste are usually compressed with small in-drum compactors in 200 l metal drums. Super compactors are used for drum pelletization, where the drum and its contents are crushed into a pellet. Incinerator ashes and

dewatered spent ion exchange resins are also treated using high-force compactors [26]. Waste such as exhausted filters, supporting ventilation equipment, metal plates, concrete, wood, and equipment—motors, centrifuges, valves, laboratory apparatus, etc.—are first disassembled, cut and/or shredded to accommodate them in drums to be super compacted or conditioned in cement.

Compaction and incineration are usually used in conjunction with a shredding process at commercial waste treatment facilities to achieve the maximum volume reduction of waste intended for long-term storage or disposal.

Spent and disused sealed sources are typically conditioned in a cement matrix inside concrete lined metal drums, concrete or steel boxes, to form a waste package suitable for transportation, storage and/or disposal. Long-lived disused sources—e.g., $^{226}$Ra, $^{241}$Am-Be and $^{226}$Ra-Be neutron sources—are encapsulated in specially designed stainless steel capsules, and hydrogenous materials are included in the package prepared for storage. Large $^{60}$Co and $^{137}$Cs sources are currently conditioned within their original shielded containers; such conditioning is limited to the preparation of sources for long-term storage [27].

Volume reduction of liquid waste is achieved by combining common methods for chemical treatment, evaporation and ion exchange/sorption of aqueous waste streams. These processes are mostly adaptations of known methods for water treatment.

Chemical treatment includes coagulation, flocculation and sedimentation of suspended radioactive solids using, for example, metal hydroxides, oxalates and phosphates. Chemical treatment is often used as a pretreatment procedure to improve the decontamination factor at ion exchange or evaporation subsequent stages. Resulted decontamination factor is about 10–100 for beta–gamma radionuclides and 1000 for alpha emitters, with a VRF of 10–100 if a small volume of wet sludge is obtained as secondary waste. If the sludge is dried by an ultrafiltration or centrifugation additional process, a VRF of 200–10,000 can be obtained [11]. The supernatant liquid should reach the clearance level of radionuclide activity to be recycled or released as a common effluent. Otherwise, the liquid stream continues to a next treatment stage.

The ion exchange technology has proven to be reliable and effective in the treatment of radioactive liquids in the nuclear industry for many years. Since the process involves the exchange of mobile ions from the liquid stream solution for ions that are electrostatically bound to the functional groups contained within a solid matrix, ion exchange/sorption procedures are appropriate when the competing salt and suspended solids content in the waste stream is low. Also, the waste should contain only very small amounts of organic contaminants. The ion exchanger—i.e., the solid matrix—can be natural or synthetic, organic or inorganic and, depending on the type of the functional group, cationic or anionic. It can be used as a packed bed in a column, or in vessels under pressure and, for small scale applications, in batch processes [28].

Most commercial ion exchangers are synthetic organic resins that can be used and regenerated repeatedly to its original ionic form, using an appropriate acid or basic solution to replace the bound contaminant ions. Inorganic ion exchangers are

almost entirely used only once. Decontamination factors range from 10 to 10,000 with an average of 100–1000, and VRFs from 500 to 10,000 [11]. Effluents from ion exchangers can be recycled or released provided that clearance levels are met. Spent resins and highly acidic and caustic radioactive liquids are obtained as secondary waste.

Evaporation is a well-established process capable of giving high decontamination factors in the order of $10^4$–$10^6$ and large volume reductions [11]. Condensate resulting from evaporation can be subsequently treated by ion exchange before it is discharged or recycled. Concentrate obtained as secondary waste can be dried to produce a salt cake or be conditioned for storage or disposal. The presence of some organic compounds that can produce explosions during evaporation, its higher cost, corrosion, scaling and foam formation are limitations to be considered when choosing this treatment option.

Since the volume of organic liquid waste—scintillation liquids containing $^3H$ and $^{14}C$, solvents, oils and miscellaneous biological fluids—from the production and use of radionuclides in nuclear research centers and in medical and industrial applications is generally small, this waste can be treated by wet oxidation, acid digestion, electrochemical oxidation, or distillation at the same laboratories using simple bench top equipment under a fume hood.

Wet oxidation involves injecting an oxidizing agent, such as hydrogen peroxide, activated sodium persulfate, ozone, Fenton's Reagent (hydrogen peroxide with an iron catalyst) or other oxidant to destroy the organic compounds. The results are carbon dioxide, water, and oxygen as well as minor concentrations of nontoxic ions, salts, and acids. The electrochemical oxidation uses, e.g., Ag(II) in a solution of silver nitrate and nitric acid, and involves placing the solution in the anode compartment of an electrochemical cell and passing through a current. This process is capable of mineralizing the organics into carbon dioxide and water completely, without emission of any toxic materials [11, 29]. Distillation is the same process known for thousands of years of heating and cooling, to separate the waste into a residue that can then be destroyed by incineration, and a recovered solvent distillate that can be reused.

An alternative to incineration that has been used in the United States is the molten salt oxidation process; a robust thermal treatment process for destroying organic waste, which consists in introducing air and waste into a bed of molten carbonate salts at temperatures between 700 and 950 °C, thus flamelessly oxidizing the organic components within the salt bath and converting the waste into $CO_2$, $N_2$, and $H_2O$ [25, 28]

## 14.6 Waste Conditioning

Conditioning may include immobilization, but it is not reduced to it. While immobilization is focused on the waste, i.e., how to convert the waste into a passively safe form, conditioning goes further to include the whole package. Immobilization

is thus the process of solidification, embedding or encapsulation to convert the waste into a passively safe waste form [1]. A passively safe waste form should be a form in which the radioactive waste may stay chemically and physically stable for hundreds or thousands of years. Waste immobilization processes place the solid or liquid waste in a container and then immobilize—encapsulate—it within a suitable matrix, or mix the liquid waste—sludge, slurry—with an immobilization matrix and pour the mix into an adequate container for solidification.

Conditioning, in its turn, involves the set of operations to produce a waste package suitable for handling, transportation, storage and disposal [1]. These operations, which can include the conversion of the waste into a solid form by immobilization, also include the waste form enclosure within a suitable container, and, if necessary, an overpack; the objective is to provide a number of necessary protective barriers against physical and chemical agents, that will reduce the potential for migration or dispersion of radionuclides for long periods of time, with a minimum need for control and safety systems, maintenance, monitoring, and human intrusion [30].

Suitability for handling, transportation, storage, and disposal is defined by specific requirements called waste acceptance criteria (WAC); which are established by the waste disposal facility operator, and are about the minimum characteristics that each waste package and its internal barriers—absorbing materials, liners, etc.—should meet to be accepted for transportation, storage and/or disposal.

### *14.6.1 Immobilization Processes*

Matrix selection for immobilization is usually ruled by the waste radiation level (LLW, ILW, and HLW), its chemical composition, and also the acceptance criteria (WAC) for storage and disposal of the disposal facility to which the waste is consigned. Important properties of immobilized waste forms are:

- Leachability,
- Chemical stability,
- Compressive strength,
- Radiation resistance,
- Thermal stability,
- Solubility,
- Noncorrosive to containers, no free liquid,
- Long shelf life, and
- Resistance to biodegradation

Radiation and thermal stability, and leach resistance, are critical requirements for immobilization of high-level waste.

Since cementation technology is well known, economic and available, it has been widely used for immobilization of low and intermediate-level radioactive waste (solid and liquid) for many years. The resulting waste form possesses good radiation

and thermal stability, high density to provide for waste shielding, good impact and compressive strength to allow for stacking, reasonable chemical stability, and moderate radionuclide leachability and waste loading. Cement is also compatible with many wastes. The main disadvantage of cement is the volume increase.

To immobilize—encapsulate—solid waste with cement, the waste is generally placed in a container, e.g., a 200 l drum, and the grouting mix is added and allowed to set. If the waste is a sludge, slurry or liquid, it is usually mixed with cement inside the container—in-drum mixing—and left to set [11]. Most of cementation techniques use Portland cement as the primary binder, but other binders, including fly ash, blast furnace slag, bentonite, zeolite and other clay materials, can be used to improve the cement mechanical performance or the retention of radionuclides [26].

Bitumen—asphalt—is a black mixture of high molecular weight hydrocarbons obtained naturally or as a residue from petroleum refining. By bituminization, waste is embedded in the molten bitumen and becomes encapsulated when the bitumen cools [26]. The primary equipment is either a multiple screw extruder or a wiped thin film evaporator [11]. Bituminization is a hot process—the equipment, bitumen storage tanks and feed lines need to be heated—which allows drying off waste streams, such as evaporator concentrates, spent ion exchange resins, filtration sludge and precipitation sludge, and greatly reduces the final waste form volume. Bituminization has high waste loading capacity and good mixability; the main disadvantages are the swelling of the product due to water uptake and generation of gases by radiolytic and microbial degradation [31].

Polymer processes do not really solidify the waste; instead, the waste is immobilized by encapsulation within the long chained molecules of the organic polymer. Until now, despise these processes were studied 30–40 years ago, they have only been used to a limited extent, mainly to immobilize ion exchange resins, and to enhance the encapsulation and penetration of the cement system into the interstitial spaces of solid waste using a polymer based grout, such as molten polyethylene, or sulfur cement.[1] Polymeric matrices that have been studied include both thermoplastic and thermosetting polymers[2]—vinylester styrene, polyethylene, polystyrene and copolymers, polyester resins, epoxy resins, urea formaldehyde resins, polyurethane, etc. All polymers present lower radiation stability than cement [28].

Over the last 15 years, a new generation of high-tech polymers has appeared in the market. The N-series of NOCHAR, Inc. was introduced in 1999 into the U.S. nuclear weapons complex as an alternative methodology for the treatment of problematic liquid radioactive waste [32]. Currently, solidified waste using NOCHAR polymers has been accepted for disposal at DOE's Handford LLW Disposal Facility, the Waste Isolation Pilot Plant (WPP), Envirocare, and Nevada Test Site (NTS). The new polymer N960 has the ability to absorb aqueous waste up to 100 times its own weight,

[1]Sulfur cements are plasticized sulfur-based cements with excellent resistance to high concentrations of nonoxidizing acids such as sulfuric, hydrochloric and phosphoric acids.

[2]Thermosetting polymers cross-link together during the curing process to form an irreversible chemical bond. Thermoplastic polymers become soft—or melt—with the application of heat, but the curing process is completely reversible because no chemical bonding takes place.

while the N910 is appropriate for the solidification of organic waste. NOCHAR polymers have also being considered for the solidification process of Cadarache LOR (Liquides Organiques Radioactifs) waste streams in France [33], and for the immobilization and solidification of complex liquid radioactive waste at the V.G. Khlopin Radium Institute (St. Petersburg, Russia) [34].

Glass and ceramic waste forms resulting from vitrification processes are the most stable forms. Vitrification can handle a wide variety of waste with large volume reductions and is a good option for the solidification of high-level wastes. High temperatures applied to melt and convert materials destroy any organic constituents with very few byproducts. Vitrification processes operate at temperatures ranging from 1100 to 3000 °C, depending on the waste composition and glass forming additives used. Resulting glass/ceramic waste form is relatively strong, highly durable and stable in corrosive environments, and leach resistant. Vitrification is currently the most widely used technology for the treatment of high-level radioactive wastes (HLW) throughout the world [35].

### *14.6.2 Packaging*

As mentioned before, a radioactive waste package is the product of conditioning, i.e., the waste form, which can be an immobilized or freely packed loose waste, together with its container and internal barriers—e.g., absorbing materials and liner—prepared according to the requirements for handling, transport, storage and/or disposal, that is, WAC [36].

The container is designed to contain, physically protect, and/or radiologically shield the waste form. It also provides a barrier to prevent water from contacting the waste form. Containers can be designed as well for in-package mixing the waste with the immobilization matrix; for grouting buoyant materials; for receiving another inner container, etc. Some containers are provided with additional internal barriers—e.g., absorbing materials and liners compatible with the waste form and the container materials—to improve the container long-term integrity and to accommodate a wider range of waste. The waste container can also be finally placed inside an overpack or canister to create the package.

Containers can be made of materials such as galvanized carbon steel, stainless steel, fiber-reinforced concrete, high-performance duplex alloys, and high-density polyethylene (HDPE), or some combination of these. Just to mention some examples, common carbon steel containers are used for storage and disposal of short-lived compacted waste and solidified waste, as sacrificial drums for palletization, or as a primary metallic container that is then placed inside an overpack or canister. Containers made of corrosion-resistant steel alloys, fiber-reinforced concrete, or polymer-coated metals, are typically intended for packaging long-lived low and intermediate-level waste (class B and class C in United States). Containers for burial are known as High Integrity Containers (HICs) and are designed to meet structural stability requirements, i.e., maintain their physical dimensions and form

under the expected disposal conditions, such as weight of overburden and compaction equipment, the presence of moisture, and microbial activity, and internal factors such as radiation effects and chemical changes, and to provide 300 years of waste containment; HIC also satisfies the transport requirements of Type A containers [18].

Since the product of conditioning—the waste package—is a major engineered component for ensuring containment, isolation and safety, it should be designed to withstand all anticipated environmental impacts during its service life. These include long-term accumulated radiation dose (alpha, beta/gamma and neutron), thermal impact, microbial activity, and chemical or corrosive attack [36]. For the purposes of transportation, storage and disposal, the waste package may also be provided with an additional canister or overpack, e.g., a heavily shielded cask.

A standard waste package for ILW, the 500 l Liquor Drum variant, designed for immobilized liquid, sludge and slurry waste forms, and incorporating a disposable paddle for in-drum mixing the waste and the immobilization medium, is shown in Fig. 14.4 [37]. This technology is available from the UK Nuclear Decommissioning Authority (NDA). Other UK NDA standard unshielded waste packages are: the 3 $m^3$ Box, for solid waste, and the 3 $m^3$ Drum, for the conditioning of sludge and resins. Drums and boxes are manufactured from austenitic to grade 316 L stainless steel with standardized lifting features.

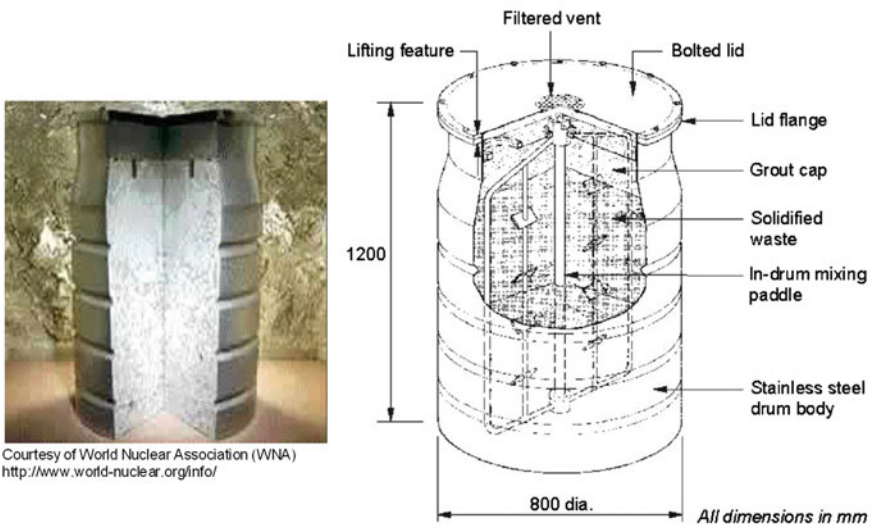


**Fig. 14.4** UK NDA standardized waste packaging [Reprinted with NDA permission]

## 14.7 Storage and Disposal

Storage and disposal are the last steps in the management of radioactive waste. They are both aimed at isolating the waste from the accessible biosphere by passive engineered and natural barriers, but differ in the intention of retrieval. While disposal means no intention of retrieval—which does not mean that is not possible—storage refers to the retention of waste with such intention any time in the future [38]. Retrieval may include further waste reconditioning or repackaging.

Short or long-term[3] interim storage is generally to allow for radioactive decay—total or partial—before release or disposal. In the case of heat generating waste, e.g., spent fuel, interim storage is also intended to reduce heat energy and prevent heat to adversely affect the future disposal system. Conditioned LLW waste, ILW waste and HLW waste—including spent sealed sources—are generally designed for long-term storage.

When designing a storage or disposal facility, specific site selection and facility construction requirements and assessments to prevent potential human disruptions, as a result of normal activities—home construction, farming, road building, mining, well drilling, etc.—and unauthorized intrusion, have to be taken into account. Such requirements and assessments are also intended to provide assurance of the proper control of waste, personnel and the surroundings, including performance of waste isolation barriers for as long as the facility will subsist. They also may include options for retrieving the waste after its emplacement, if this is considered to be appropriate.

Commonly accepted disposal options are: (a) near-surface disposal, with or without engineered barriers, at ground level or at depths of tens of meters below ground level, and (b) deep geological disposal at depths between 250 and 1000 m for repositories, or 2000–5000 m for boreholes.

Very low radioactive waste, such as soil and rubble arising from decommissioning activities, as well as mining and mineral processing waste that arise in large volume, are typically disposed of in near-surface facilities similar to conventional landfills, but with institutional surveillance and control of the facility and measures to stabilize the waste in situ. Waste is usually disposed of at ground surface covered with various layers of rock and soil, but if the activity concentration of long-lived radionuclides is above the limitation, it has to be disposed of below the ground surface.

Currently, the majority—$\sim$90 % by volume—of LLW around the world is sent to engineered near-surface disposal facilities. Many long-term waste storage options for ILW and HLW are being studied worldwide too; they seek to provide public accepted, safe, and environmentally sound solutions to the management of radioactive waste [39]. Some existing and planned disposal and storage facilities are briefly described next.

[3]Long-term in radioactive waste disposal refers to periods of time which exceed the time during which active institutional control can be expected to last (approximately 300 years).

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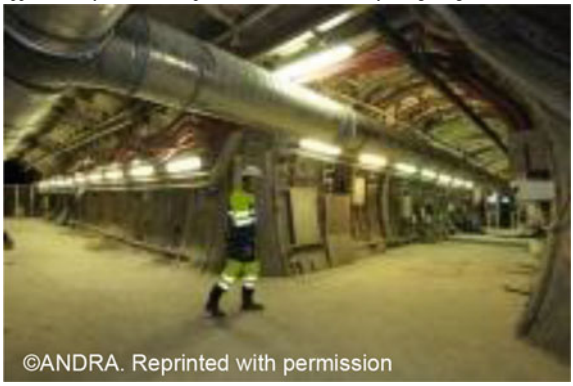
| *France* | ANDRA, the French National Agency for Radioactive Waste Management, controls two near-surface repositories: (1) Center Aube with 1,000,000 $m^3$ capacity, in operation since 1992 (LILW), and (2) the Cires disposal facility with 650,000 $m^3$ capacity, operating since 2003 (VLLW) . Both facilities are shallow repositories within a low permeability clay layer at a depth of about 15 meters. Centre Aube uses concrete vaults in a box design with mobile roof for the storage of concrete containers, stainless steel drums (100, 200, 400 and 800 l), and metallic boxes (5 $m^3$ and 10 $m^3$), filled with short-lived low- and intermediate-level waste (LILW SL) conditioned by generators using standard procedures. Very low level waste (VLLW) is sent to the Cires facility located in Morvilliers, licensed to dispose of very-low-level waste. Processed waste, packaged in metal containers, drums, plastic-lined big bags, etc., is disposed of straight above ground, on a plastic impermeable membrane settled in large trenches excavated to a depth of a few meters in the clay layer. When filled, trenches are backfilled with sand, and covered with a HDPE membrane and a clay layer [40].<br>  <br>©ANDRA Cires disposal facility. Reprinted with permission<br>  <br>©ANDRA Center Aube. Reprinted with permissio<br>As an alternative strategy to the long-term surface storage, France authorized the construction of an underground research laboratory in Meuse–Haute Marne for the deep geological disposal of high level (HLW) and intermediate long-lived waste (ILW LL). Since 2004, many experiments have been conducted to study the claystone formation and its behavior to confirm its feasibility. In its drifts network, at 500 m deep approximately, it has been possible to observe directly the geological formation in real time [41].<br>  <br>©ANDRA. Reprinted with permission<br>The future disposal facility is being designed to be reversible for at least 100 years. This means that an early definitive closure and the retrieval of packages are expected, if necessary. |
| *Germany* | Currently, there is no repository operating in Germany, although the former salt mines Asse and Morsleben were used for low- and intermediate-level radioactive waste (LILW) disposal for more than 20 years. Both repositories, at present planned for decommissioning and final closure due to problems of mechanical stability and brine intrusion, are in Permian salt formations. The storage of radioactive waste in Asse was largely carried out in chambers at 750 m depth. ILW was stored at 511 m. Since it is considered that the mine is seriously damaged, today's recommended option is the complete retrieval of the stored radioactive waste and its movement to an alternative location. The Morsleben repository has two shafts to provide access to a system of tunnels, caverns, pits and connection halls opened between 320 and 630 m below the ground level. Different techniques were employed for disposal: stacking, in situ solidification of liquid waste using lignite filter ash as a binding agent, and dumping of solid wastes. The option recommended for the closure of Morsleben is the sealing of major disposal areas, and an extensive backfilling with salt concrete to provide mechanical support and reduce the cavities [42].<br>Meanwhile, dry interim storage facilities at NPPs, transport cask storage facilities (TBL)—e.g., Gorleben, Ahaus—and other above ground facilities, have been used for the storage of spent nuclear fuel, vitrified HLW, residues from reprocessing, and non-heat-generating waste from nuclear power plants, medicine, research and industry. |

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| | Other two sites for deep geological disposal have been studied: a never mined salt dome at Gorleben for the disposal of HLW, and the former iron mine Konrad, for the disposal of waste from hospitals, research centers and industry (LILW). An exploratory mine was installed at Gorleben and was operating until 2013. The site is now kept at a minimum until the new site selection procedure not rules it out. Konrad is an abandoned iron ore mine that is currently converted into a deep repository for radioactive waste with negligible heat generation. Compared with other iron ore mines, Konrad is exceptionally dry, and the covering layer of clay rocks, which is up to 400 m thick, assures its sealing against groundwater. Konrad is authorized to dispose up to ~300,000 $m^3$ of LILW in chambers dug between the 800 m level and the 850 m level. Three types of standard packages will be used: cylindrical concrete containers, cylindrical cast iron cask, and rectangular high volume containers. It is important to note that Germany classifies radioactive waste only in two categories: heat-generating waste and non-heat-generating waste, and that disposal of short-lived LLW and ILW on or near the surface were never considered [43]. |
| *Japan* | The Low-Level Radioactive Waste Disposal Center at Rokkasho-Mura, a near-surface repository at a depth less than 50 m in a clay host rock, has been operating for more than 20 years as the only final repository for low-level waste generated at nuclear power plants throughout Japan.<br>  <br><br>The total approved capacity is 80,000 $m^3$, planned to be expanded to 600,000 $m^3$. Waste is classified by activity level and origin in L3, L2, L1, and TRU (matching VLLW, LLW, ILW and TRU respectively). Vitrified waste is considered HLW. The existing repository uses a trench disposal method for VLLW (L3), mainly containing short-lived radionuclides, and concrete vaults with engineered barriers for LLW (L2) in two disposal sites, one for homogeneously solidified waste, and the second for solidified dry active waste [44].<br>  <br><br>A subsurface, tunnel type facility, at an intermediate depth of 50–100 m, has also been considered at Rohhashu for the disposal of ILW (L1) containing long-lived radionuclides. The engineered barrier system includes the host rock, a concrete soil backfill, a layer of bentonite as low permeability material, and a reinforced concrete vault. A test cavern for studies associated with such disposal facility, e.g., of geological structure, faults and fractures, groundwater chemistry, rock mechanics, etc., was excavated at 100 m below the ground surface.<br>High-level waste is planned for disposal in a stable geological formation at a depth of more than 300 m. The concept is to emplace the vitrified waste canisters, encapsulated in strong metal containers (overpacks) surrounded by a compacted bentonite clay buffer, in underground tunnels [45]. Studies are being carried out in underground shafts for crystalline rock and sedimentary rock. |

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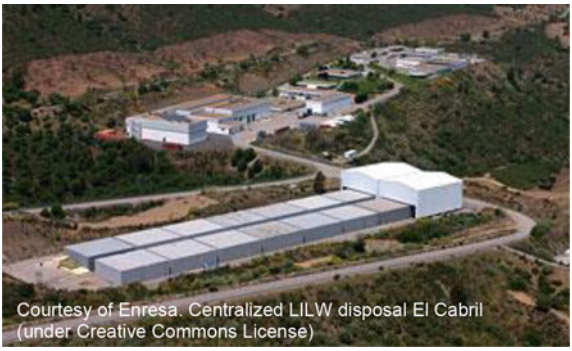
| *Spain* | The centralized disposal facility El Cabril, in Hornachuelos, Córdoba, is in operation since 1992. It is a vault type near-surface disposal facility for LILW with a total internal capacity of more than 100,000 $m^3$. The disposal system uses natural and engineered barriers to isolate waste for a decay time of 300 years. The multibarrier system includes first the metal package containing the immobilized waste or pellets; following, a concrete overpack with a thick cap where metal packages are reconditioned by backfilling the void spaces with grout; and finally, a reinforced concrete disposal vault. When the vault is completely filled, it is closed with a reinforced concrete closing slab and weatherproofed. The disposal concept incorporates the potential retrievability of waste packages [46].<br>  <br><br>A new platform with four disposal cells for VLLW, with a total capacity of 130,000 $m^3$, has been recently added to the centralized disposal facility El Cabril. These cells, located near the LILW zone, are designed for a period of 60 years. Each cell has an isolation artificial geological barrier made of 1 m compacted clay and 0.03 m geo bentonite. The isolation barrier also includes two layers of high density polyethylene (HDPE). Each cell is divided in longitudinal strips called lines of operation, protected by a light roof structure.<br>  <br><br>Although the deep geological disposal is the preferred option, a Centralized Interim Storage (CISF) using vault technology with passive cooling is to be constructed in Villar de Cañas, Cuenca, to store all the spent fuel from nuclear power plants and vitrified HLW produced in Spain for 60 years, by which time a repository for permanent disposal is expected to be available. |
| *Sweden* | The first component of the Swedish system for the disposal of radioactive waste is the Swedish Final Repository (SFR) near Forsmark, operated since 1988. The repository was designed for the centralized disposal of short-lived LILW from nuclear power plants and the use of radioactive materials in medicine, research and industry. |

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The repository is located in the bedrock, about 50 m below the sea. Most active waste is disposed of in a concrete silo surrounded by a clay buffer (filled with bentonite), constructed in one of the rock caverns. The other four caverns are one for LLW, two for concrete tanks with dewatered ion exchange resins, and one for ILW. The facility is connected to the ground by two parallel tunnels 1 km long [47].


The second component of the system is the Central Interim Storage Facility for Spent Nuclear Fuel (Clab) located near the Oskarshamn power plant. Here, the spent fuel from nuclear power plants is stored in one of 4 storage pools present in each two rock caverns about 30 m below the surface. To transfer the fuel, the entire transport cask is lowered into a fifth pool, where the fuel assemblies are transferred to a storage canister, before take them down to the rock caverns in a water-filled elevator cage [48].

The final repository for spent nuclear fuel in Forsmark and the encapsulation plant in Oskarshamn are part of the last component of the system. SKB applied for the license to the two facilities in 2011, after three decades of full-scale research in the Äspö Hard Rock Laboratory (HRL). The spent fuel will be placed—encapsulated—in sealed steel and copper canisters prior to its disposal in rock vaults surrounded by bentonite clay, at a depth of 500 m in the bedrock. The Äspö Laboratory is a unique research facility, at 500 m below the surface, to study the interaction of bentonite clay and copper canisters with the rock in real conditions [49]. SKB is also planning a final repository for long-lived LILW.

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| *Switzerland* | Spent fuel and vitrified HLW are packaged in containers and transported to the ZWILAG centralized interim storage in Würenlingen, where they are stored in tightly sealed transport and storage cask, in the cask storage hall.<br>   <br>LILW waste from medicine, industry, research, and nuclear power plants is also brought here to be processed. For example, combustible materials, metallic parts, concrete, and other solid matter, can be thermally decomposed or melted by a plasma process at temperatures of up to 20,000 °C, and large components from nuclear power plants can be disassembled and decontaminated. In the storage building for ILW, the 200 l steel drums filled with conditioned waste are stacked one on top of the other in storage shafts; individual storage shafts are covered with solid concrete lids. All operations are remote controlled [50].<br>The Swiss waste management concept foresees two deep repositories: one for HLW and one for LILW, using disposal tunnels or caverns depending on the waste types. After several years of investigations, the National Cooperative for the Disposal of Radioactive Waste (NAGRA) proposed two geological siting regions—Zürich Nordost and Jura Ost—to be further investigated for the emplacement of a HLW and LILW repository or a combined repository. Opalinus clay is the only host rock under consideration. Switzerland also has two rock laboratories:<br>•The Grimsel Test Site (Canton Bern), established in 1984 as a center for underground research and development, and<br>•The Mont Terri rock laboratory (Canton Jura), established in 1998 for the specific study of argillaceous rock, such as opalinus clay, as a host rock<br>The Grimsel Test Site, with approximate 1,000 m of tunnels and 5,000 m of cored boreholes, is located at an altitude of 1,730 m above sea level, in the granitic formations of the Aar Massif; it presents ideal conditions for investigating the functioning of both the geological and the engineered barriers of deep repositories. The Mont Terri rock laboratory, 300 m underground and with 600 m of galleries and niches to carry out experiments, is an international research platform to provide essential knowledge about the safety and feasibility of deep geological disposal [51]. |
| *UK* | The UK national LLW disposal facility is operating since 1959 at Drigg, in West Cumbria. It is a near-surface repository with engineered concrete vaults. The site also has seven old trenches now prepared for closure. This repository receives waste—compacted and containerized—from educational, medical and research establishments, and the nuclear and nonnuclear industry for their disposal. The waste arrives at the facility in large metal ISO freight containers and is transferred to the Grouting Facility, where each container is filled with grout to create a solid structure, previously to be placed in the vault [52].<br>  <br>The UK strategy for decommissioning at Dounreay, the famous Britain's center for experimental fast breeder research, includes: (1) the disposal of VLLW—mostly rubble and soil—to landfill; (2) the decontamination of metal LLW for recycling; and (3) the disposal of LLW to a long-term (50–100 years) interim storage at the site. The LLW interim storage at Dounreay consists of up to six concrete lined and roofed vaults, just below |

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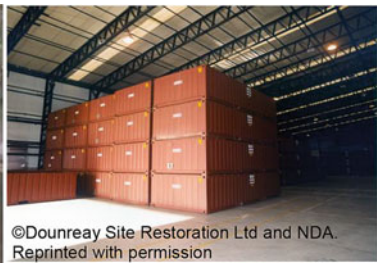
surface, with a total capacity of 175,000 $m^3$ of solid waste. Mild steel drums of encapsulated waste are placed in a large metal ISO freight half height container and grouted with a cementitious material before being put in the vault. Once the vault is full, it is backfilled with grout to create a monolithic block [53].


ILW is packaged using the standardized waste packaging mentioned earlier. There are several packaging plants operating at Sellafield, Dounreay, Harwell, Trawsfynydd, Windscale and Winfrith. Packaged ILW are sent to the Magnox highly engineered interim storage facility (ISF) at Berkeley site, designed to minimize maintenance, while assuring an appropriate monitoring and control [54]. The waste will be stored there inside transportable, self-shielded cast iron containers, until a geological disposal facility is available. The ISF at Berkeley is the largest of its kind in the UK and will hold up to 1004 ILW containers.


HLW is processed and stored at Sellafield. The Sellafield site, operational since the 1940s, has been home to a number of facilities, including the Magnox and Thermal Oxide (THORP) reprocessing plants, and the Sellafield MOX fuel plant. Sellafield HLW process involves vitrification in borosilicate glass, and decay and cooling in an up-to-date engineered store—the Vitrified Product Store—for at least 50 years before final disposal [55].


UK's policy is to dispose of higher activity radioactive waste in deep geological facilities, except in Scotland, where the policy is for long-term management in near-surface facilities.

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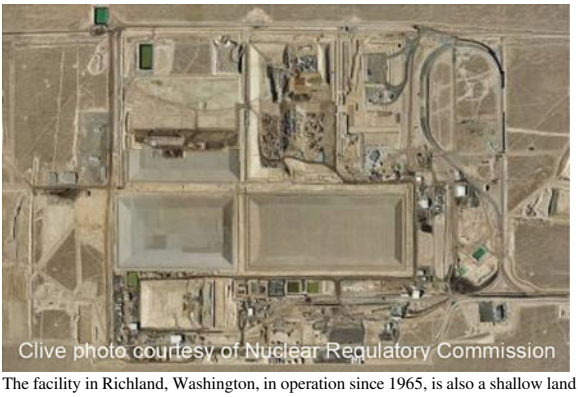
| *USA* | There are four active commercial disposal sites in the U.S. that accept LLRW, located in the states of South Carolina, Washington, Utah, and Texas. The LLW shallow land burial at Barnwell, South Carolina, is the only commercial facility that has operated without interruption since 1971. The disposal is designed to contain Class A, B, and C radioactive waste for a period of 300 years. Barnwell receives waste only from Connecticut, New Jersey, and South Carolina. Sealed waste containers are placed in concrete rectangular or cylindrical vaults located in trenches excavated in clay-rich soil.<br>  <br><br>When a vault is full, it is closed with a concrete lid. Sandy clay is added as backfill material to fill voids between vaults. Finally, the completed trench area is covered with an engineered cap consisting of a compacted clay layer, a geosynthetic clay liner, a high density polyethylene (HDPE) liner, a sand layer and a sandy topsoil layer [56].<br>Clive, UT, an above grade disposal facility, accepts waste from all regions of the United States. The disposal is limited to Class A, Mixed Waste and NORM. This site, located 80 miles west of Salt Lake City, operates both bulk and containerized disposal.<br>  <br><br>The facility in Richland, Washington, in operation since 1965, is also a shallow land burial facility of packaged waste into unlined trenches. It receives low-level Class A, B, and C radioactive waste from 11 states in the Northwest and Rocky Mountain compacts, as well as accelerator produced radioactive material (NARM)—e.g., pipe scale from oil and gas pipelines, radium sources, smoke detectors, exit signs, and electron tubes, etc.—from all states. The facility covers 100 acres of land located near the center of the Department of Energy's Hanford nuclear site [57]. |

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The new Waste Control Specialists LLR (WCS) facility in Andrew, Texas, is designed for the long-term disposal of commercial and federal Class A, B and C LLW, and low-level mixed waste. This engineered facility is excavated in natural red bed clay to an over 30 m depth, and covered with a 2 m thick liner system including a 30 cm thick layer of reinforced concrete and a geosynthetic layer. The waste is placed in steel reinforced concrete containers and capped for disposal. The facility is also authorized for the storage and processing of greater than Class C waste, sealed sources, transuranic waste (TRU) , and byproduct material [58].


Spent fuel is stored at NPPs in specially designed pools or in airtight steel cylinders called dry cask storage systems. WCS is also seeking a license for a spent fuel interim storage facility (ISF) at Andrews's site. The spent fuel would be stored for a period of 60-100 years housed in steel-reinforced concrete. Other HLW—alpha mixed waste, TRU, fissile waste from research and development activities using enriched uranium, etc.—are stored at DOE facilities: the Idaho National Engineering and Environmental Laboratory (since 1952); the Oak Ridge National Laboratory (since 1973); Los Alamos National Laboratory (since 1957); the Hanford site (since 1940), and the Savannah River burial site (from 1970 to 1995).

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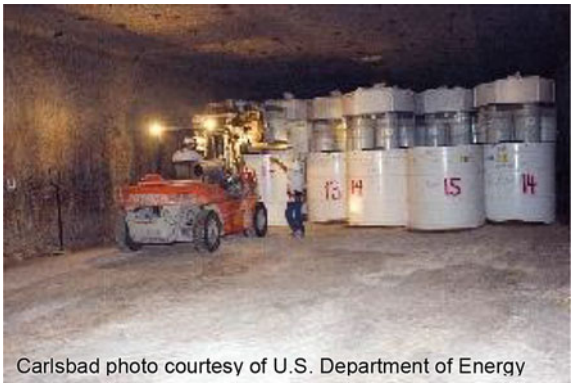


A final repository for HLW long-term storage is projected at two sites. The first site is the Waste Isolation Pilot Plant (WIPP) in Carlsbad, New Mexico. This is a deep geologic repository that, since 1999, is accepting TRU waste generated from defense activities for its permanent disposal. The waste is stored in a salt dome at a depth of 655 m in specially mined disposal rooms [59]. The second is the Yucca Mountain site located on federally owned desert land in the state of Nevada, which was designated in the 1980s for the deep geological disposal of spent fuel and other HLW. Its current status remains uncertain. The repository is projected as an underground mine in the unsaturated zone—at about 300 m above the water table—with a complex of tunnels occupying approximately 8.1 $km^2$ at a depth between 300 and 1,200 m below the surface. The host rock is made from compacted volcanic ash formed more than 13 million years ago. The design should allow for retrieval.

## 14.8 Management and Disposal of Disused Sealed Sources

A sealed source became a disused source if, for example: (1) it is spent, i.e., its activity decays to a level unsuitable for the original purpose; (2) the experiment, program or practice authorized to use the source ends; (3) there is a fault in the source or the equipment in which it works, e.g., the source is leaking, bent, corroded, cracked or badly scratched, etc.; or it is stuck in the equipment; (4) the equipment holding the source is declared obsolete, e.g., a moisture/density gauge with old electronics is replaced for a new one, an apparat using a $^{137}$Cs chloride source is replaced by other source with an alternative form of cesium, etc.; (5) the source is not traceable to any authorized holder (an orphan source).

Because of the nature of the radionuclides contained in sealed sources—energy, activity, and half-life—spent and disused sources can represent a hazard and must be transferred to, or seized by, an authorized organization for reuse, conditioning or disposal.

The terms spent source and disused source have been indistinctly used in the meaning that the source is no longer in use. When the last happens, it has occurred that some radioactive sources have been left behind and then, found or stolen by people unaware of what they were dealing with. If a high activity sealed source, either in a device or container, is abandoned, disposed of improperly, lost or stolen, the device or container can be broken into pieces, the source can be taken out of its shielding, the source encapsulation can become smashed, or the entire apparatus can end up melted down with a scrap, and contaminate individuals, materials and the environment; in addition to causing radiation injuries—erythema, tissue damage, amputation and even death—to the individuals finding or handling the source or observers, due to excessive exposure [27]. Unfortunately, this has been learned from the radiological incidents and accidents, including fatalities, which have been reported over the last 30 years.

Management options for spent and disused sources are shown in Fig. 14.5. They include:

- Returning the spent source to the supplier whenever possible or transferring it to another authorized user. The supplier is usually a specialized manufacturer with all the conditions to manage the sources; thus, the best solution is to pursue a return agreement with the source purchase. If the user does not have a return agreement, the source can be transferred to another user for a different application, provided that all the information related to the source ownership and status is in order and switched through the regulatory authority. For instance, a $^{137}$Cs source declared spent at a hospital may be given to a nearby university for use as calibration source. The new user must be an authorized user, the source should be declared spent or disused, and a leak test and activity update should be performed. It is important to note that a leaking source must always be taken out of service and handled as radioactive waste;

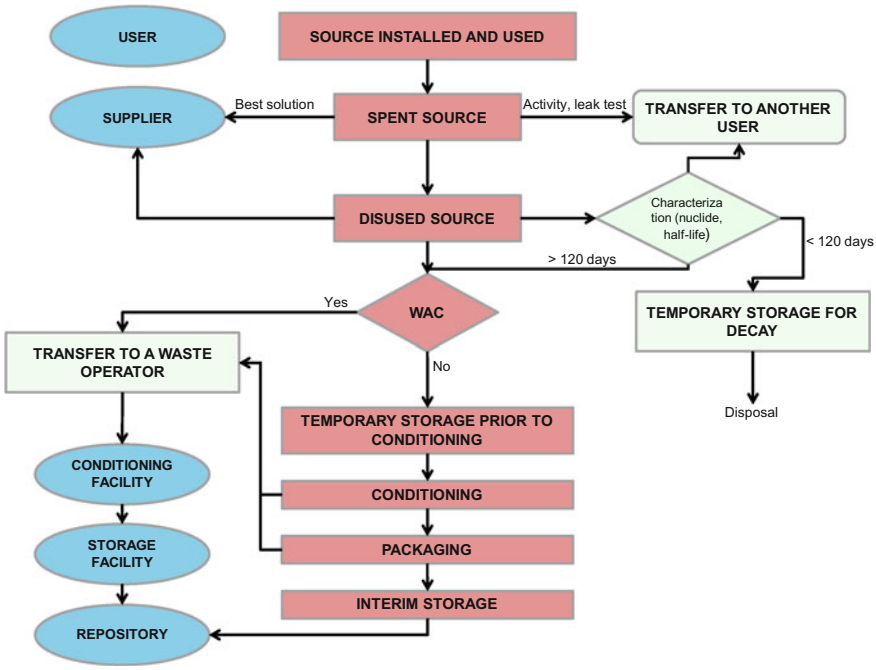


**Fig. 14.5** Spent source management options

- Temporary storing the spent or disused source for decay on site and disposing of it after reaching the clearance level if the half-life of the source is 120 days or less, e.g., $^{32}$P (14.3 days), $^{125}$I (59.4 days), $^{192}$Ir (74 days). This option is subject to a suitable decay store, well developed and implemented administrative procedures, and an adequate management system;
- If there is no reuse or return to supplier options, and the source is not of short-lived radionuclides, the best practice is to declare it as waste. If it meets the waste acceptance criteria (WAC) , the source should be transferred at the earliest opportunity to a radioactive waste operator for storage or disposal. Technology for conditioning and/or packaging may be necessary on site to meet the WAC; also, an interim short-term storage prior to send it as waste to a repository. If there are no conditions on site, a little or no delay transfer to a waste operator for processing and disposal is favored.

It is a requirement that spent or disused sources that may pose hazards to inadvertent intruders should not be placed in near-surface disposal facilities. The preferred disposal method for a high activity spent source is shaft or borehole disposal [27]. Boreholes can be readily drilled offshore as well as onshore, in host rocks both crystalline and sedimentary. This capability significantly expands the range of locations that can be considered for disposal and is an attractive proposition for sealed radioactive sources from medical and industrial applications.

Borehole construction and site characterization are comparative easy, and make this method particularly suitable.

The disposal in borehole facilities falls between the two well-established options of disposal in near-surface facilities and disposal in geological facilities. The concept entails the emplacement of disused sealed radioactive sources, surrounded by an encapsulation matrix and placed in durable containers, in an engineered facility, bored or drilled, and operated directly from the surface. The depth can vary depending on the sources, engineered barriers and geological characteristics of the site. Other engineered barrier is the borehole backfill, which could include cement, bentonite slurry, and a loose fill of bentonite granules or sand. This type of disposal could be economic while minimizing the probability of human intrusion. Siting the facility away from known mineral and water resources decrease even more the likelihood of human intrusion.

## 14.9 Conditioning of Spent Sealed Sources

Conditioning of spent sources, as well as of any other radioactive waste, involves the operations to produce a waste package acceptable for safe handling, storage, transportation and final disposal; in compliance with the relevant WAC, and that, after storage, still could be safely retrieved and transported. Typically, such package is designed to reduce the migration of radionuclides for long periods—up to 300 years—to avoid a potential intruder exposure and provide an extra confinement for leaking sources.

The method for conditioning depends on the disused source characteristics—type of radiation, activity, half-life, and chemical toxicity—and the technology used for handling, transport and storage. To comply with WAC, disused sources may need to be removed from their original shipping/storage containers and then reconditioned and repackaged.

To increase storage efficiency, short-lived disused sources can be placed in appropriate sized steel drums, and steel or concrete boxes, closed with a lid of the same material. This allows for stacking sources, which are in irregularly shaped containers, and also maintain the source retrievability for further disposal. Depending on the space, surface dose rate and total activity, more than one source can be accommodated in the same recipient. This is recommended for operational convenience in interim storages for short-lived sources, during the period of radioactive decay prior to clearance. Description of the sources including radionuclides, activities, manufacturing information, dimensions and geometry, shielding details, results and date of leak test, measured dose rate, and gross weight should be engraved on the recipient.

Neutron sources can be conditioned using the pipe overpack containers S-300 and S-100, approved as standard DOT 7A Type A packages. Both containers employ high density polyethylene neutron shielding inside the stainless steel pipe, which reduces external neutron dose equivalent rates per unit activity in the

**Fig. 14.6** S-100 pipe overpack container (Photo courtesy of Los Alamos National Laboratory)

container. This allows container loading with greater activity content, while maintaining compliance with external dose rate limitations for both packaging and disposal. The S300 pipe overpack consists of a 30 cm diameter pipe positioned within a 200 l drum by means of fiberboard/plywood fillers and internal neutron shielding materials. The pipe component body, lid and bolt flange are constructed of stainless steel. A butyl rubber or ethylene propylene O-ring is required for pipe component closure. The S-100 container shown in Fig. 14.6 consists of a 15 cm diameter pipe component positioned within a 200 l drum by means of neutron shielding materials. The pipe component body, lid and bolt flange are also constructed of stainless steel [60].

There is a well proven method for the conditioning of $^{226}$Ra needles, tubes, and applicators used as brachytherapy sources. As shown in Fig. 14.7, the method includes encapsulation in prefabricated small size stainless steel capsules filled with a lead-based alloy to facilitate their retrieval, and the emplacement of several welded capsules inside a stainless steel cylinder, after sealing and testing the capsules for leak tightness. Steel cylinders are then placed inside a lead container for shielding and conditioning in a 200 l mild steel drum with concrete lining. This package could be filled later with cement mortar to totally immobilize the sources or it could be opened to retrieve them for future final conditioning.

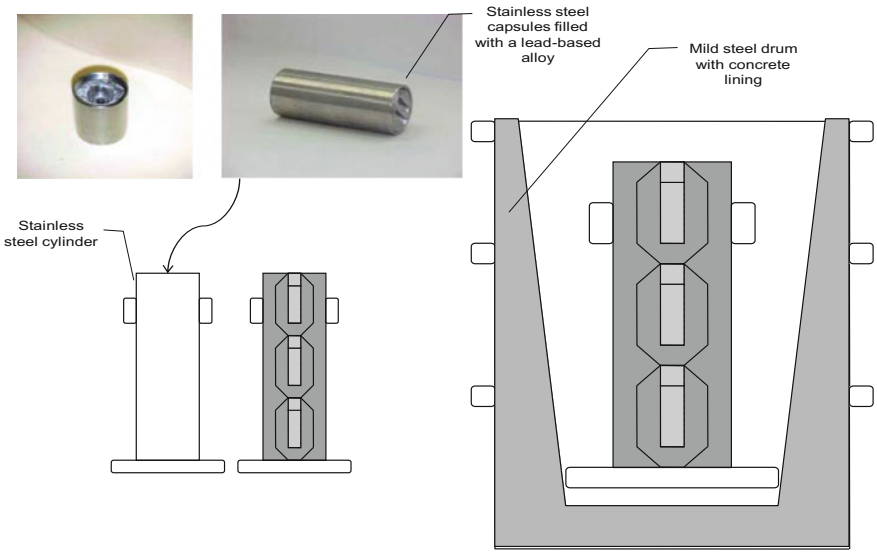


**Fig. 14.7** Radium sources conditioning technology

All relevant information on the conditioned sources—radioisotope, date of conditioning, number of capsules, total activity, etc.—need to be engraved in metal plates attached to the packages.

High activity and long-lived gamma disused sources, e.g., $^{60}$Co or $^{137}$Cs radiotherapy or irradiation sources, are usually part of heavy shielded devices that are not suitable for conditioning by any of the above-mentioned methods. Also, the handling and conditioning of such sources requires hot cells and remote/slave manipulators. If they cannot be returned to the supplier, they should be held in their respective transport or transfer containers, if any, on secure sites and under control to prevent intrusion, pending further solutions for processing or disposal. A temporary measure or a measure only to be taken in special circumstances is the removal of the source holder from the equipment. The source holder can then be placed in a suitable container or the void space of an appropriate capacity drum fixed inside a larger drum with concrete, or in concrete or steel boxes that have larger volumes and loads than drums.

## References

1. International Atomic Energy Agency (2003) Radioactive waste management glossary 2003 edition. IAEA, Vienna
2. International Atomic Energy (2004) Application of the concepts of exclusion, exemption and clearance, Safety Standards Series No. RS-G-1.7. IAEA, Vienna
3. NRC Regulations Title 10, code of federal regulations, Part 20, standards for protection against radiation, 25 July 2013

4. U.S.NRC, "§ 30.71 Schedule B," United States Nuclear Regulatory Commission (2015). http://www.nrc.gov/reading-rm/doc-collections/cfr/part030/part030-0071.html
5. International Atomic Energy (2014) radiation protection and safety of radiation sources: international basic safety standards, IAEA Safety Standards Series GSR Part 3. IAEA, Vienna
6. U.S.NRC (2003) NUREG-1640. Radiological assessments for clearance of materials from nuclear facilities. U.S. Nuclear Regulatory Commission, Washington
7. International Atomic Energy Agency Safety Guide No.GS-G-3.3 (2008) The management system for the processing, handling and storage of radioactive waste. IAEA, Vienna
8. International Atomic Energy Agency (2009) Predisposal management of radioactive waste GSR Part 5. IAEA, Vienna
9. International Atomic Energy Agency (1995) Principles of radioactive waste management. Safety fundamentals. IAEA, Vienna
10. International Atomic Energy Agency (2007) Identification of radioactive sources and devices, IAEA Nuclear Security Series No. 5. IAEA, Vienna
11. International Atomic Energy Agency (2001) Handling and processing of radioactive waste from nuclear applications. Technical Report Series No. 402. IAEA, Vienna
12. International Atomic Energy Agency (2014) Treatment of radioactive gaseous waste. IAEA-TECDOC-1744. IAEA, Vienna
13. International Atomic Energy Agency (2009) Classification of Radioactive Wastes Safety Guide No. GSG-1. IAEA, Vienna
14. NRC Regulations Title 10, Code of Federal Regulations, Part 35, General Domestic Licenses for Byproduct Material, 1 Oct 2007 and 23 July 2008
15. Committee on Improving Practices for Regulating and Managing Low-Activity Radioactive Wastes, National Research Council, Improving the Regulation and Management of Low-Activity Radioactive Wastes, Washington: The National Academies Press, 2006
16. U.S. Nuclear Regulatory Commission (2002) Radioactive Waste: Production, Storage, Disposal (NUREG/BR-0216, Revision 2), U.S. Nuclear Regulatory Commission, U.S. Nuclear Regulatory Commission
17. NRC Regulations Title 10, Code of Federal Regulations, Part 60, Disposal of high-level radioactive wastes in geologic repositories, July 2013
18. NRC Regulations Title 10, Code of Federal Regulations, Part 61, Licensing requirements for land disposal of radioactive waste, July 2013
19. Advisory Committee on Nuclear Waste (ANCW) White Paper (2007) History and Framework of Commercial Low-Level Radioactive Wastes Management in the United States. NUREG-1853, U.S. Nuclear Regulatory Commission, Washington
20. Andrews A (2006) CRS report to congress. Radioactive waste streams: waste classification for disposal. order code RL32163. Congressional Research Service, Washington
21. U.S. NRC (2015) Byproduct material. United States Nuclear Regulatory Commission. http://www.nrc.gov/reading-rm/basic-ref/glossary/byproduct-material.html
22. U.S. Department of Energy (2015) ENERGY.GOV Manifest Information Management System. U.S. Department of Energy. http://mims.doe.gov/Default.aspx
23. EPA (2014) Resource Conservation and Recovery Act (RCRA) Subtitle C. United States Environmental Protection Agency. http://www3.epa.gov/region02/waste/csummary.htm
24. Cecille L (CEC), Kertesz C (CEA) (1990) ASHES 90, Treatment and Conditioning of Radioactive Incinerator Ashes. In: Proceedings of a technical seminar jointly organized by the Commission of the European Communities (CEC) and the Commissariat à l'Energie Atomique, Aix-en-Provence
25. Oh C-H (2001) Hazardous and radioactive waste treatment technologies handbook. CRC Press, Boca Raton
26. Ojovan MI (2011) Handbook of advanced radioactive waste conditioning technologies. Woodhead Publishing, Cambridge
27. International Atomic Energy Agency (2014) Management of disused sealed radioactive sources, IAEA Nuclear Energy Series No. NW-T-1.3. IAEA, Vienna

28. International Atomic Energy Agency (2002) Application of ion exchange processes for the treatment of radioactive waste and management of spent ion exchangers. Technical Reports Series No. 408. IAEA, Vienna
29. Valdovinos V, Monroy-Guzman F, Bustos E (2014) Treatment methods for radioactive wastes and its electrochemical applications. In: Dra. Hernandez Soriano MC (ed) Environmental risk assessment of soil contamination. Tech-Open Access under CC BY 3.0 license
30. Bayliss C, Langley K (2003) Nuclear decommissioning, waste management, and environmental site remediation, 1st edn. Butterworth-Heinemann, Burlington
31. Sneyers A, Volckaert G, Neerdael B (2001) The Belgian research, development and demonstration program on the geological disposal of long-lived and high-level radioactive waste and spent fuel in a clay formation: status and trends. In: Proceedings of the WM-01 conference, Feb 25–March 1, Tucson, AZ
32. Nochar (2015) N-Series Nuclear Products," Nochar. Inc.http://nochar.com/services-for-individuals/
33. Vaudey C-E et al (2013) NOCHAR polymers: an aqueous and organic liquid solidification process for Cadarache LOR (Liquides Organiques Radioactifs). In: WM2013 Conference, Feb 26–28, Phoenix
34. Pokhitonov Y, Kelley D (2013) The polymers for liquid radioactive waste solidification: a lost chapter in the history of engineering or a step forward?. In: WM2013 Conference, Feb 24–28, Phoenix, 2013
35. Ojovan MI, Lee WE (2013) An introduction to nuclear waste immobilisation, 2nd edn. Elsevier
36. International Atomic Energy Agency (2006) Development of specifications for radioactive waste packages, IAEA-TECDOC-1515. IAEA, Vienna
37. United Kingdom Nirex Limited (2007) Generic Repository Studies. Generic Waste Package Specification Volume 2—Justification. Nirex Report no. N/104 (Issue 2)," Nirex
38. International Atomic Energy Agency (2011) Disposal of radioactive waste, specific safety requirements no. SSR-5. IAEA, Vienna
39. World Nuclear Association (2015) Storage and disposal options, World Nuclear Association. http://www.world-nuclear.org/info/nuclear-fuel-cycle/nuclear-wastes/appendices/radioactive-waste-management-appendix-2–storage-and-disposal-options/
40. ANDRA (2013) French national radioactive waste management agency, The waste disposal facility in the Aube District, ANDRA
41. ANDRA (2010) French National Radioactive Waste Management Agency, The presence of Andra in the Meuse and Haute-Marne districts., ANDRA
42. Bundesamt für Strahlenschutz (2015) NUCLEAR WASTE MANAGEMENT. Transport - Storage - Surveillance of Radioactive Waste. BfS, The Federal Office for Radiation Protection. http://www.bfs.de/EN/topics/nwm/repositories/repositories_node.html
43. Bundesamt für Strahlenschutz (2008) Konrad repository. The Federal Office for Radiation Protection, Science creates trust, Salzgitte
44. JNFL (2015) Japan Nuclear Fuel Limited, Corporate Profile, Japan Nuclear Fuel Limited
45. NUMO (2008) Nuclear Waste Management Organization of Japan, Geological Disposal of radioactive Waste in Japan, Nuclear Waste Management Organization of Japan
46. Zuloaga P (2014) HLW storage and SL-LLW disposal in Spain. In: 19th REFORM Group Meeting, Salzburg, 1–5 Sept 2014
47. Svensk Kärnbränslehantering AB, "SKB. The Final Repository SFR," SKB (Swedish Nuclear Fuel and Waste Management Company), 2015. http://www.skb.com/our-operations/sfr/
48. Swedish Nuclear Fuel and Waste Management Co (2010) Clab, Stockholm: SKB
49. Swedish Nuclear Fuel and Waste Management Co (2006) The Äspö Hard Rock Laboratory. SKB, Stockholm
50. ZIWLAG (2014) Installations and Operational Processes, Würenlingen: Zwilag Zwischenlager Würenlingen AG
51. NAGRA (2014) Annual Report, National cooperative for the disposal of radioactive waste, Wettingen

52. LLW Repository Ltd (2015) National Repository, LLW Repository Ltd. http://llwrsite.com/national-repository/
53. Dounreay Site Restoration Ltd. (2016) Waste, Dounreay Site Restoration Ltd. http://www.dounreay.com/waste
54. TEKLA Potential (2012) Magnox ILW Interim Storage Facility, Berkeley, Trimble Company. http://www.tekla.com/uk/bim-awards-2012/magnox-storage-facility.html
55. Nuclear Decommissioning Authority (NDA) (2009) UK Radioactive Higher Activity Waste Storage Review, NDA
56. South Carolina Department of Health and Environmental Control (2007) Commercial Low-Level Radioactive Waste Disposal In South Carolina. South Carolina Department of Health and Environmental Control, Columbia
57. Washington State Department of Health (2004) Final Environmental Impact Statement, Commercial Low-Level Radioactive Waste Disposal Site. DOH Publication 320-031, DOH, 2004
58. Waste Control Specialists (WCS) Our Facilities, Waste Control Specialists LLC, 2016. Waste Control Specialists LLC
59. U.S. Department of Energy (2015) Waste Isolation Pilot Plant, The U.S. Department of Energy. http://www.wipp.energy.gov/index.htm
60. Los Alamos National Laboratory (2011) S300 Container and S100 Container," Los Alamos National Security, LLC. http://osrp.lanl.gov/s300.shtml and http://osrp.lanl.gov/s100.shtml

# Chapter 15
# Transport of Radioactive Materials

To protect the public and the environment, there are internationally agreed regulations for the safe transport of radioactive materials[1] that prescribe technical requirements to limit the external radiation and contamination from packages, assure the containment of its radioactive content, and prevent criticality and excessive heat during transport. Regulations for the safe transport of radioactive material [1] are based on the Fundamental Safety Principles [2] and the International Basic Standards [3], and are the basis for the General Provisions Concerning Class 7—Radioactive Material—of the UN Recommendations on the Transport of Dangerous Goods [4]. The regulations for the safe transport of radioactive materials apply to transport by land (road or rail), by air, and by water (sea, rivers, lakes, etc.).

Radiation safety during transport largely depends on the package design and operation. Package means the packaging, i.e., the components or materials for containment and safety, and its contents—the radioactive material—prepared for transport. Package designs and certain consignments[2] require approval by the regulatory authority. The NRC and the Department of Transportation (DOT) share responsibility as primary regulators for the safe transport of radioactive materials in the United States. NRC establishes the requirements for packaging, preparation for shipment and transportation of licensed material, fissile material, and a quantity of other licensed material in excess of Type A quantity (10 CFR Part 71) [5], while DOT covers all aspects of transportation, including specifications for packaging, specifications for tank cars, shipper's responsibility and certification, special handling requirements, and labeling, marking, and placarding (49 CFR Parts 171-180) [6]. The Postal Service has jurisdiction over accepting only very small quantities of

[1]For transportation purposes, radioactive material is any material which activity concentration is greater than that of the exempted material established by the standards.
[2]Consignment is any package or packages, or radioactive material load, presented by a solely consignor for transport.



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radioactive materials for domestic mail, falling within the category of excepted package or excepted instrument [7].

According to mentioned regulations, there are three types of carriers: (a) common, (b) contract, and (c) private. Common and contract carriers usually provide transport services to others and are not licensed by NRC or an Agreement State. In these cases, the responsibility for safety rests with the consignor (shipper), who prepares the package, assures the corresponding labeling and marking, and attaches the required documentation. The carrier responsibility is usually limited to the vehicle condition, driver training, and certain operational requirements stated by the consignor. Private carriers, in contrast, own the radioactive material which they carry and are licensed by NRC or an Agreement State. Examples of private carriers who transport their sources from one jobsite to another are: industrial radiographers, portable gauge users, and well loggers. Radionuclide producers may also deliver their own radiopharmaceuticals to nuclear medicine clinics.

## 15.1 Classification of Material and Packages

$A_1$ and $A_2$ are the activity limits used to determine the type of packaging required for a particular radioactive material consignment; $A_1$ is for special form material and $A_2$ for other than special form, called normal form radioactive material. $A_1$ and $A_2$ values, in TBq, along with activity concentration limits for exempt material, in Bq/g, and activity limits for exempt consignments, in Bq, can be found in Table 2 of IAEA transport regulations [1] and in Table Appendix A to 10 CFR Part 71 [5].

$A_1$ and $A_2$ are also measurements of the radionuclide radiological risk. The $A_1$ value results from worst case scenarios regarding external exposure from the unshielded source at a certain distance. Thus, $A_1$ is the activity of that radionuclide that will result in a dose rate of 100 mSv/h at a distance of 1 m. $A_2$ is the least activity resulted from the applicability of the most conservative worst case scenarios, including external exposure, external beta radiation to skin, inhalation, ingestion, and external gamma radiation from immersion in a gaseous cloud of material released from a breached package [1, 8, 9]. Some radionuclides have been assigned unlimited $A_1$ and $A_2$ values because their specific activity and toxicity are so low that, it is unlikely that a person, in the vicinity of the damage package, will receive the dose criterion, i.e., a dose of 50 mSv at a distance of 1 m for more than 30 min. Any radioactive material for which the $A_2$ value is unlimited is considered LSA-I.

According to transport regulations [1, 5, 6], materials can be classified as

- Low Specific Activity (LSA), when it is a material with low activity per unit mass, which poses little hazard even if released in an accident. LSA could be of one of three groups: LSA-I, LSA-II, and LSA-III, depending on the contents and the specific activity. Uranium and thorium ores and concentrates, and depleted uranium, are LSA-I materials; water with a tritium concentration of up to 0.8 TBq/l, and any material with an average specific activity that does not exceed

$10^{-4}$ $A_2$/g for solids and gases, and $10^{-5}$ $A_2$/g for liquids are LSA-II materials; a consolidate waste and a solid activated material, excluding powder, with an average specific activity $< 2 \times 10^{-3}$ $A_2$/g, are LSA-III materials;

- Surface contaminated object (SCO), when it is a solid object not radioactive itself, but which has radioactive material distributed on its surface because it has been contaminated. SCO could be of one of two groups: SCO-I, SCO-II, depending on non-fixed and fixed contamination levels on the accessible surfaces averaged over 300 $cm^2$;
- Special form radioactive material,[3] when it is either an indispersible solid radioactive material or a sealed capsule containing radioactive material, which can only be opened by destroying it, i.e., a sealed source tested and approved as such;
- Low dispersible radioactive material, when it is either a solid radioactive material or sealed capsule containing a solid radioactive material, which is limited dispersible and is not in powder form. The radiation level at 3 m from an unshielded low dispersible radioactive material should not exceed 10 mSv/h;
- Fissile material, when it is a material containing any of the fissile nuclides: $^{233}U$, $^{235}U$, $^{239}Pu$, and $^{241}Pu$, excluding natural uranium and depleted uranium;
- Uranium hexafluoride, when it is fissile or nonfissile uranium hexafluoride ($UF_6$) in solid form.

For transportation purposes and according to the radioactive material, a package is classified as [1, 5, 6]

- Excepted package, if it: (a) is an empty package which previously contained radioactive material; (b) contains a limited number of instruments or articles—gauges, smoke detectors, electronic apparatus or similar devices—which activity do not exceed the activity limits specified in Table 4 of regulations [1, 6]; (c) contains articles manufactured of natural uranium, depleted uranium, or natural thorium; or (d) contains radioactive material in limited quantities; generally, the quantity for solid or gaseous contents is $10^{-3}$ of that permitted in a Type A package; for liquids, the quantity is reduced to $10^{-4}$ of that allowed in a Type A package. The radiation level at any point on the external surface of an excepted package is limited to 5 μSv/h;
- Industrial package, if it contains LSA or SCO radioactive materials. There are three types of industrial packages according to the requirements for withstanding routine and normal conditions of transport: Type IP-1, Type IP-2, and Type IP-3. Industrial package safety is assured more by the nature of the contents, than by the strength of the packaging, though, Type IP-1 are used to

[3] A special form radioactive material:

- Should have at least one dimension of not less than 5 mm;
- Will not break or shatter under impact, percussion, and bending tests;
- Will not melt or disperse in heat test;
- The water activity from leaching tests will not exceed 2 kBq.

transport LSA-I and SCO-I materials; LSA-II, LSA-III, and SCO-II require Type IP-2 or Type IP-3 packages. The quantity of LSA or SCO in a single package is limited by an external radiation level of 10 mSv/h at 3 m from the unshielded material;

- Type A package, if it contains a radioactive material in special form with and activity < $A_1$ or any other radioactive material with an activity < $A_2$. A Type A package should maintain its integrity during normal transport conditions, e.g., falling from the vehicle, being dropped during manual handling, being exposed to the weather, being struck by a sharp object, or having other packages or cargo stacked on top, but is not designed to prevent the loss of the contents under accident conditions. In such case, no adverse health or environmental effects are expected due to the limited amount of radioactivity allowed in the packaging;
- Type B or Type C package, if it contains a radioactive material with an activity greater than $A_1$ or $A_2$. Type B and Type C packages require a competent authority certificate of approval. Regarding approval, Type B packages are classified as unilaterally approved—Type B(U)—if they are only approved by the competent authority of the country of origin of the design, or multilaterally approved—Type B(M)—if they have to be approved by the competent authorities of the countries through, or into which, the consignment is to be transported. Type B packages are designed to withstand impact/crush, penetration, thermal, and water immersion tests. Type C packages are required to withstand puncture/tearing, enhanced thermal, and higher impact velocity tests. Test requirements take into account a large range of accidents for land, sea, and air transport which can expose packages to severe dynamic forces. More severe accident forces in an air transport accident are taken into account by the Type C test requirements.

Fissile materials and uranium hexafluoride ($UF_6$) have additional packaging requirements. Packages containing fissile material should also be designed to remain subcritical under normal and accident transport conditions; they are classified as FISSILE, unless exempted by IAEA transport regulations [1]; packages carrying fissile uranium hexafluoride($UF_6$) , in addition to criticality, should be designed to protect against its unusual physical characteristics.

In Fig. 15.1 is shown a RH-72-B package Type B(M)F-96 [10]. This is a Type B (M) package designed to safely transport transuranic waste. F-96 indicates that the package was approved for fissile material under the 1996 edition of transport regulations. The package is a leak tight large cylinder with inner and outer containment vessels. The cylinder fits into circular impact limiters, similar to shock absorbers, designed to protect the container and its contents in the event of an accident. The RH-72B has a lead liner to shield people from gamma rays and an outer thermal shield to protect the container against potential fire damage [11].

Consistent with material classification, package type, and fissile or nonfissile characteristics, the package—container, overpack—should be assigned to one of the UN numbers specified in the Table 1 of IAEA transport regulations [1], followed by the proper shipping name. For example, the corresponding number for


**Fig. 15.1** RH- 72-B transuranic waste transportation container

package RH-72-B in Fig. 15.1 is UN 3329 and the proper shipping name is: RADIOACTIVE MATERIAL, TYPE B(M) PACKAGE, FISSILE.

## 15.2 Testing Requirements

Package compliance with transport regulations is demonstrated by testing full scale packages, scale models, or mock-ups of specific package parts; or by calculations; or with a combination of these methods. Tests are performed to specimens of LSA-III material, special form radioactive material, and low dispersible radioactive material; and to prototypes or packaging samples of industrial packages, and Type A, Type B and Type C packages, prepared, as closely as possible, to the final package presented for transport.

Leaching test is required for LSA-III and low dispersible radioactive material (LDRM). In this test, the sample is immersed for 7 days in water at ambient temperature. The water should have an initial pH of 6–8 and a maximum conductivity of 1 mS/m at 20 °C. The total activity measured in the nonabsorbed free volume—at least 10 % of the volume of the solid sample itself—at the end of the 7-day period should not be greater than 0.1 $A_2$. Low dispersible materials (LDRM) are also subject to the enhanced thermal test and the impact test at a speed no less than 90 m/s.

Specimens of special form radioactive material should pass the following tests without releasing its radioactive contents out of allowed limits:

- Impact test onto a rigid, flat target, from a height of 9 m;
- Percussion test, the specimen should be struck by the flat face of a mild steel bar, so as to cause an impact equivalent to that resulting from a free drop of 1.4 kg through 1 m;
- Heat test at a temperature of 800 °C for a period of 10 min and then cooled;
- For long, slender sources with both a length of 10 cm or greater, and a minimum length to width ratio of 10, the bending test, by striking the free end of the specimen with a steel bar, so as to produce an impact equivalent to that resulting from a free vertical drop of 1.4 kg;
- Leaching test by immersion in water at ambient temperature or a volumetric leakage assessment according to the standard ISO 9978 for sealed sources [12].

To demonstrate the ability to withstand normal conditions of transport, prototypes, and samples of industrial packages and Type A packages, undergo the water spray test, the free drop test, the stacking test, and the penetration test. In the water spray test, the prototype—sample—is subject to a water spray that simulates exposure to a rainfall of approximately 5 cm/h, for at least one hour. In the free drop test, the specimen is dropped onto a target from a height of 0.3−1.2 m depending on the package mass, so as to experience the maximum damage. The target is a flat, horizontal surface. For rectangular fiberboard or wood packages not exceeding a mass of 50 kg, a separate sample is free dropped onto each corner from a height of 0.3 m. For cylindrical fiberboard packages not exceeding a mass of 100 kg, the sample is dropped onto each of the rim quarters.

In the stacking test, the prototype or sample is caused to undergo, for a period of 24 h, a compressive load equal to the greater of the following: (a) 5 times the maximum weight of the package; (b) 13 kPa multiplied by the vertically projected area of the package. The load is uniformly applied to two opposite sides of the sample, the top and the base. In the penetration test, the specimen is placed on a rigid, flat, horizontal surface, which will not move significantly during the test. A bar with a diameter of 3.2 cm and a mass of 6 kg is dropped onto the center of the weakest part of the specimen, so that, if it penetrates sufficiently far, it will hit the containment system. The drop height should be 1 m. An additional free drop test from 9 m and an additional penetration test with a drop height of 1.7 m are required for Type A packages designed to contain liquids and gases.

Neither loss nor dispersal of the radioactive contents, and no more than 20 % increase in the maximum radiation level at any external surface of the package, should occur from these tests.

To get an idea of the barriers included in the simplest packages for the transport of radioactive materials, Fig. 15.2 provides an example of a typical containment system for non-sealed radioactive sources in Type A packages.

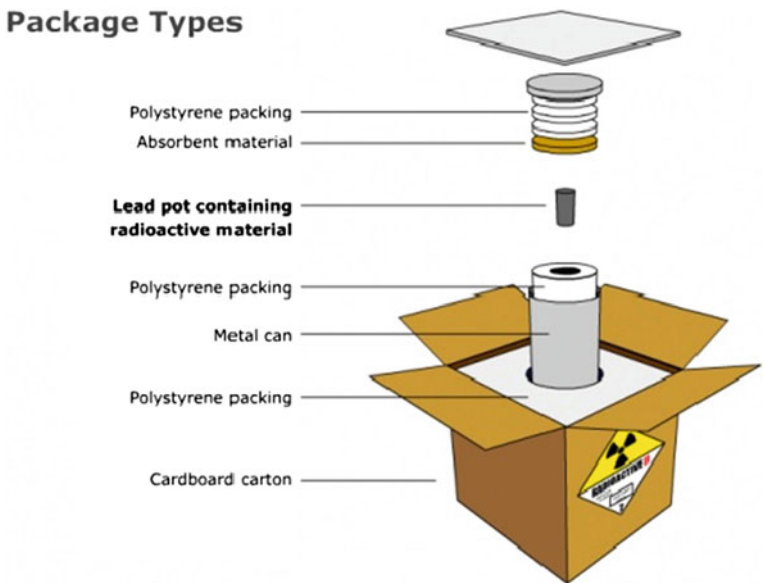


**Fig. 15.2** Type A package [Courtesy of the UK Health Protection Agency]

To demonstrate the ability to withstand severe transport accident conditions, prototypes, and samples of Type B and Type C packages are subject to the cumulative effects of a mechanical test, a thermal test, and a water immersion test (in this order), designed to cause the maximum damage. The mechanical test and the thermal test are applied sequentially to the same specimen. A separate sample or prototype is used for the immersion test.

The mechanical test consists of three different drop tests. For drop I—impact—the sample or prototype is dropped onto the target from a distance of 9 m. For drop II—penetration—the same sample or prototype is then dropped onto a bar rigidly mounted perpendicularly on the target. The drop height is 1 m measured from the impact point to the upper surface of the bar. For drop III—crush—the sample or prototype undergoes a dynamic crush by dropping a solid mild steel plate with a mass of 500 kg from 9 m onto it. The drop height is measured from the underside of the plate to the highest sample point.

Following the mechanical test, the same specimen is exposed for a period of 30 min, to a liquid hydrocarbon–air fire with an average temperature of at least 800 °C, fully engulfing the specimen. The sample should not be artificially cooled after the fire, and any combustion of materials of the specimen should be permitted to proceed naturally. The fire test duration for Type C packages, as for an aircraft accident, is set at 60 min.

In the water immersion test, a separate sample is immersed under a head of water of at least 15 m for a period no less than 8 h, in a position which leads to maximum damage. Type B(U) and Type B(M) packages containing more than $10^5$ $A_2$ and

Type C packages should be subject to an enhanced water immersion test, where the sample is immersed under a head of water of at least 200 m for no less than 1 h.

Fissile materials should be transported so as to maintain subcriticality during all transport conditions. Packages containing fissile materials are hence subject to the cumulative effects of the mechanical test with penetration (drop II) and either impact (drop I) or crush (drop III), and the thermal test before undergoing an immersion test under a head of water of at least 0.9 m for no less than 8 h. This sequence is chosen to provide conditions which will allow the free access of water into the package, together with damage which could rearrange the fissile contents.

If the mass of $UF_6$ is 0.1 kg or more, an hydrostatic test is necessary to demonstrate that no cylinder tearing, deformation or major failure occurs when it is pressurized with water up to 2.76 MPa (1.38 MPa at least for multilateral approval design) [13].

No important loss of the containment system integrity for the radioactive material within the package should occur from these tests. The shielding retention for all packages and the maintenance of subcriticality for packages containing fissile materials should also be taken into account when evaluating testing results.

Table 15.1 briefly describes the testing requirements associated with each package type and transport conditions. It also provides a few examples of materials transported in the particular package type.

## 15.3 Limits and Categories

The radiation level in the vicinity of a package, overpack or freight container, or an unpackaged LSA-I or SCO-I is indicated by the Transport Index (TI) [1]. The TI is determined as the maximum radiation level, in mSv/h, measured at a distance of 1 m from the external surfaces of the package, overpack, freight container or unpackaged LSA-I and SCO-I; multiplied by 100. The resulting value is rounded up to the first decimal place, except that a value of 0.05 or less may be considered as zero. TI is dimensionless.

TI, along with the surface radiation level, is then used to

(a) Determine the appropriate category for the package or overpack;
(b) Determine if the package or overpack should be transported under exclusive use;
(c) Apply the spacing requirements during in-transit storage operations;
(d) Apply the requirements for special arrangements; and
(e) Identify the number of packages allowed in a freight container or conveyance.[4]

[4]A vehicle, vessel, hold, compartment, or defined deck area of a vessel, or aircraft.

**Table 15.1** Summary of testing requirements

| Package type | Transport condition | Testing requirements | Examples of materials |
|---|---|---|---|
| Excepted | Routine transport | Vibration testing is required | Empty containers, smoke detectors |
| IP-2 | Normal transport, including minor mishaps | Free drop<br>Stacking or compression | LLW, ores and ore concentrates (uranium and thorium), tritiated water with an activity < 0.8 TBq/l |
| IP-3 | Normal transport, including minor mishaps | Water spray test<br>Free drop test<br>Stacking or compression<br>Penetration | |
| Type A | Normal transport, including minor mishaps | Water spray test<br>Free drop test<br>Corner drop test<br>Stacking or compression<br>Penetration | Pharmaceuticals, technetium generators, certain sealed sources |
| Type B | Severe accident | Cumulative effects of free drop with impact, penetration and crush<br>Thermal test<br>Water immersion test<br>Enhanced water immersion test if activity > $10^5 A_2$ | Industrial radiography devices, spent fuel, HLW, $^{60}Co$ radiotherapy or irradiation sources |
| Type C | Severe accident | Cumulative effects of free drop with impact and crush<br>Enhanced thermal test<br>Puncture-tearing test<br>Impact test<br>Enhanced water immersion test | Small amounts of high-activity materials transported by aircraft |
| Fissile material package | Severe accident | Cumulative effects of free drop with penetration and either impact or crush<br>Thermal test<br>Submersion test<br>Hydrostatic test for $UF_6$ | |

Exclusive use means the sole use of a conveyance, or a large freight container, by a single consignor. If so, all initial, intermediate, and loading and unloading operations, as well as shipment, are carried out under the direction of the consignor or consignee.

The consignor, consignee, and carrier should have a radiation protection program, commensurate to the operations they carry out. The program should at least ensure the appropriate personnel training in radiation protection, all the required documentation, including approvals and certificates, the provisions to respond to an emergency during the transport of radioactive materials, and the resources to conduct the transportation safely.

According to the maximum radiation level at any point on an external surface and TI, packages—overpacks and freight containers—are assigned to one of three categories. In addition, each package—overpack and freight container—carrying fissile material, other than excepted fissile material, should be labeled as such. The labels corresponding to these categories are shown in Fig. 15.3; also the label for fissile materials

- WHITE-I when the maximum surface radiation level is less than 0.005 mSv/h and TI is zero;
- YELLOW-II when the maximum radiation level is more than 0.005 mSv/h, but less than 0.5 mSv/h and TI is more than zero, but less than 1;
- YELLOW-III when the maximum radiation level is more than 0.5 mSv/h, but less than 2 mSv/h and TI is more than 1, but less than 10.

Both the TI and the maximum surface radiation level are to be considered when determining the category. In case TI satisfies the condition for one category and the maximum surface radiation level, the condition for a different category; the package—overpack or freight container—must be assigned to the higher category. For example, if TI = 0.5 (YELLOW-II), but the maximum radiation level is 0.65 mSv/h, the package should be assigned to YELLOW-III category. The TI of an overpack, freight container or conveyance, with more than one package inside, is determined as either the sum of the TIs of all packages contained in it, or by directly measuring the radiation level.

Consignments under special arrangements—i.e., consignments that do not conform with all the provisions and ought to be specially approved by the competent authority—are always assigned to category YELLOW-III no matter how low the radiation level is. Although, if the maximum surface radiation level is > 2 mSv/h, the package, overpack or freight container must also be transported under exclusive use. 10 mSv/h is the maximum radiation level on the external surface of a package, overpack or freight container transported under exclusive use.

Another important requirement is the limit of non-fixed contamination[5] on the external surfaces of any package, which should be kept as low as practicable. Under routine transport conditions, non-fixed contamination averaged over 300 $cm^2$—or over the whole area of the surface if it less than 300 $cm^2$—should not exceed 4 $Bq/cm^2$ for beta and gamma emitters, and low toxicity alpha emitters, or 0.4 $Bq/cm^2$ for all other alpha emitters [1]. Same limits apply to the external and internal surfaces of overpacks, freight containers, tanks, intermediate bulk containers, and conveyances. Internal surfaces are exempt from measurement if overpacks, freight containers, tanks, intermediate bulk containers, and conveyances are dedicated to the transport of unpackaged radioactive material under exclusive use.

[5]Non-fixed contamination is the contamination that can be removed from a surface during routine transport conditions.

Packages, overpacks or freight containers carrying fissile material have a criticality safety index (CSI) assigned. This number is used to provide control over the accumulation of packages, overpacks or freight containers carrying fissile material. The CSI can be obtained by dividing 50 by a number N (CSI = 50/N). The number N is such, that a package array based on it would be subcritical, both under normal and accident transport conditions. The CSI for a package, overpack or freight container is also rounded up to the first decimal place.

According to IAEA transport regulations [1], the number N is evaluated using tentative N numbers. For example, an array of five times N packages can be tested under normal transport conditions to see if it is subcritical, and an array of two times N packages can be tested separately under accident conditions to see if it is subcritical. The smaller of these two values is then used to determine the CSI. Standards for arrays of fissile material packages are found in 10 CFR §71.59 [5] and in Appendix VI of IAEA transport regulations [9]. However, as stated by 49 CFR §172.403 [6], the CSI is that assigned in the NRC or DOE package design approval, or in the certificate of approval for special arrangement, or the certificate of approval for the package design issued by the competent authority for import and export shipments.

Except for consignments under exclusive use, the maximum TI of any package or overpack is 10. Likewise, the maximum CSI is 50.


**Fig. 15.3** Class 7 category labels for radioactive materials

Regarding dose rate, except for packages or overpacks transported under exclusive use, or under exclusive use and special arrangement, the maximum radiation level at any point on the external surface of any package or overpack is 2 mSv/h. The maximum radiation level on any external surface under exclusive use is 10 mSv/h.

## 15.4 Marking, Labeling and Placarding

Each package—container, overpack—should bear the name or address of either the consignor or the consignee, or have attached the shipping document containing this information, and the corresponding UN number, package type and category. The package in Fig. 15.4 is a Type A package, containing a radioactive material UN 2915, i.e., in nonspecial form, and nonfissile or fissile-excepted. Note the shipping documents attached. An overpack must bear in addition the word OVERPACK on the exterior.

The package in Fig. 15.4 is also a DOT-7A design, which does not require the approval of either DOT or NRC, for domestic shipment or for international transportation of nonfissile radioactive material. The DOT specification 7A is the only authorized Type A package in the DOT regulations based totally on performance test conditions.


**Fig. 15.4** A labeled Type A container with radioactive material

Each package—container, overpack—should also carry the category label with a minimum size of 100 × 100 mm affixed to two opposite sides. The information on the category label includes TI, radionuclide content and activity. If multiple radionuclides are present, the most restrictive should be listed, i.e., those that represent 95 % of the hazard present. The package in Fig. 15.4 has a Category YELLOW-II label and a TI = 0.2.

If the package—container, overpack—contains fissile material, the label for criticality safety shown in Fig. 15.3 is also used. Fissile label should bear the mass —in grams—of the fissile material in place of the activity, and the CSI stated in the certificate of approval, instead of the TI.

Additionally and in agreement with DOT regulations [6], radioactive materials are subject to the following package marking requirements:

- Gross weight if > 50 kg;
- "TYPE IP-1", "TYPE IP-2", "TYPE IP-3", "TYPE A" "TYPE B(U)", or "TYPE B(M)," as appropriate to the package;
- For each IP-1, IP-2, IP-3, or Type A package, the design country of origin code; USA for United States;
- For each DOT 7A Type A packaging:
  - "USA DOT 7A Type A";
  - Name of packaging manufacturer (the person certifying that the package meets all requirements for a Type A package);
- For Type B packages, the basic ionizing radiation symbol—the trefoil radiation symbol shown in Chap. 12—resistant to the effects of fire and water, plainly marked by embossing or stamping (not on a sticky label);
- For Type B and fissile material packages, the applicable DOT, NRC or DOE package certificate ID number, as specified in the relevant certificate, e.g., USA/9212/B(M)F-96;
- Exclusive use domestic transportation of LSA materials and SCO is excepted from other marking requirements, but must be stenciled or marked as "RADIOACTIVE-LSA" or "RADIOACTIVE-SCO," as appropriate;
- Excepted packages are excluded from other marking requirements, but must be marked with the UN number for the material and be labeled as "Radioactive Material" on the inside or outside of the pack.

Placards are used on large freight containers and tanks, as well as on road and rail vehicles, carrying consignments under exclusive use. As shown in Fig. 15.5, these are labels very similar to that of package categories, but bigger, with a minimum size of 250 × 250 mm, to facilitate its reading even at a certain distance. They are affixed to each side wall and to each end wall of a large freight container or tank, and to each side of a vehicle.

Placards are required if any radioactive material package in the freight container or tank, or on the vehicle, bears the YELLOW-III label, or if the shipment includes LSA and SCO to be consigned as exclusive use, e.g., LSA or SCO shipped in


**Fig. 15.5** Placard

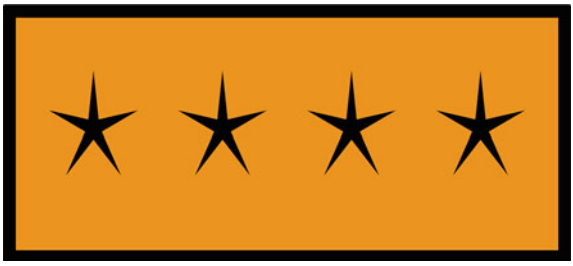

**Fig. 15.6** Placard for UN number

excepted packaging, liquid LSA-I material, or unpackaged LSA material or SCO. If the radioactive material is from a single UN number, the number can be displayed in the lower half of the placard or in a separate placard with an orange background color and a black border, as shown in Fig. 15.6. The symbols **** denotes the places for the UN number to be displayed.

49 CFR §172.600 [6] also requires shippers to provide emergency response information on hazardous materials shipments. Shipments of excepted radioactive material packages—packages containing limited quantities, instruments or articles, or Empty packages—are not subject to the emergency response information requirements.

The emergency response information must provide at least

- A basic description and name of the radioactive material;
- Immediate precautions to be taken in the event of an accident or incident, including methods for handling fires, and spills or leaks in the absence of fire, and
- Preliminary first aid measures.

The consigner should also provide an emergency response telephone number which must be monitored on a 24-h basis while the shipment is in transportation.

In addition, 49 CFR 172.204 [6] requires shippers to provide a certification statement that the hazardous material offered for transportation is in compliance with all applicable regulations. The following declaration can be used: "I hereby declare that the contents of this consignment are fully and accurately described above by the proper shipping name, and are classified, packaged, marked and labeled/placarded, and are in all respects in proper condition for transport according to applicable international and national governmental regulations."

## References

1. International Atomic Energy Agency (2012) Regulations for the safe transport of radioactive material 2012 Edition, IAEA Safety Standards Series No. SSR-6. IAEA, Vienna
2. International Atomic Energy Agency (2006) Fundamental safety principles. IAEA, Vienna
3. International Atomic Energy (2014) Radiation protection and safety of radiation sources: international basic safety standards, IAEA Safety Standards Series GSR Part 3. IAEA, Vienna
4. United Nations (2013) Recommendations on the transport of dangerous goods, model regulations, ST/SG/AC.10/1/Rev.18. Geneva, UN, New York
5. 10 CFR Part 71 (1995) Packaging and transportation of radioactive material. U.S. Nuclear Regulatory Commission
6. 49 CFR (2004) Subtitle B, Chapter I, Subchapter C, Hazardous Materials Regulations. U.S. Department of Transportation
7. US Postal Service Publication 52 (2013) Hazardous, restricted, and perishable mail, July 2013
8. U.S. Department of Transportation (1998) Radioactive material regulations review RAMREG 001-98. DOT Research and Special Programs
9. International Atomic Energy Agency (2014) Advisory material for the iaea regulations for the safe transport of radioactive material (2012 Edition), No. SSG-26. IAEA, Vienna
10. Sampson M (2013) Certificate of compliance no. 9212 for the model no. RH-TRU 72-B PACKAGE. Docket No. 71-9212. TAC No. L24419. Division of Spent Fuel Storage and Transportation, Office of Nuclear Material Safety and Safeguards, Rockville
11. Waste Isolation Pilot Plant, Transuranic waste transportation containers. U.S. Department of Energy, Carlsbad
12. International Organization for Standardization (ISO) (1992) Radiation protection—Sealed radioactive sources—Leakage test methods, ISO 9978: 1992(E). ISO, Geneva
13. Institute American National Standards (2001) American national standard for nuclear materials —Uranium Hexafluoride—Packaging for Transport ANSI N14.1-2001. Revision of ANSI N14.1-1995. American National Standards Institute Inc, Washington

# Chapter 16
# Emergency Exposure Situations

The assessment of the probability, magnitude and consequences of potential exposures, and, therefore, the introduction of the corresponding engineering protections, is a requirement for the activities and facilities where radiation sources are used. It is also a requirement that emergency plans be prepared to deal with and mitigate the consequences of such potential exposures should any of them occur. Emergency plans are reviewed and improved on a regular basis as well, to demonstrate their functionality and suitability.

## 16.1 Potential Exposure

Potential exposure is an exposure that is not expected to occur with certainty, i.e., its probability of occurrence is < 1, but that could result from an anticipated operational occurrence, accident, specific event or sequence of events, including equipment failure, human error, or even natural disasters—e.g., hurricanes, earthquakes, tsunamis, and floods—malicious events, and inadvertent human intrusion into disposal sites [1].

Such events cannot be predicted in detail, but they can be theoretically anticipated; i.e., what can occur, and the relevant consequences if it does, can at least be predicted.

Even though events leading to potential exposures in some applications can be such that affect only few people, the consequences can be severe for them. While they do not have the implications of an accident in a nuclear power plant, they do have occurred with alarming frequency and devastating effects for those involved.

Here are some examples of foreseeable scenarios that may give rise to potential exposures in most common applications:

- An industrial radiography source failing to retract correctly to its shielded position



H. Domenech, *Radiation Safety*, DOI 10.1007/978-3-319-42671-6_16

- An unsafe entry into an irradiation room with the source in open position
- A fire breaching the integrity of a sealed source
- A fire damaging the shielding of a waste package
- A dropped or detached industrial radiography source
- An inadvertent spillage of radioactive material

## 16.2 The Analysis of Potential Exposures

The analysis of potential exposures starts with constructing and evaluating scenarios, a technique which principles are well known and often used in engineering [2]. The aim behind the scenarios is to describe the different paths of development —sequences that may lead to the exposure—through the observation of certain postulated key factors or initiating events, such as a system or device failure, human error, negligence, and so on. The estimated probability and magnitude of the exposure are hence compared to previously established constraints and criteria [3]. The key steps for the analysis are summarized in Fig. 16.1.

Since the logic of events allows for the reasoning of those events and how they relate to each other via, among other things, a well-founded causal ordering, the first step is to be familiar with the specifics of the facility, source or activity; the

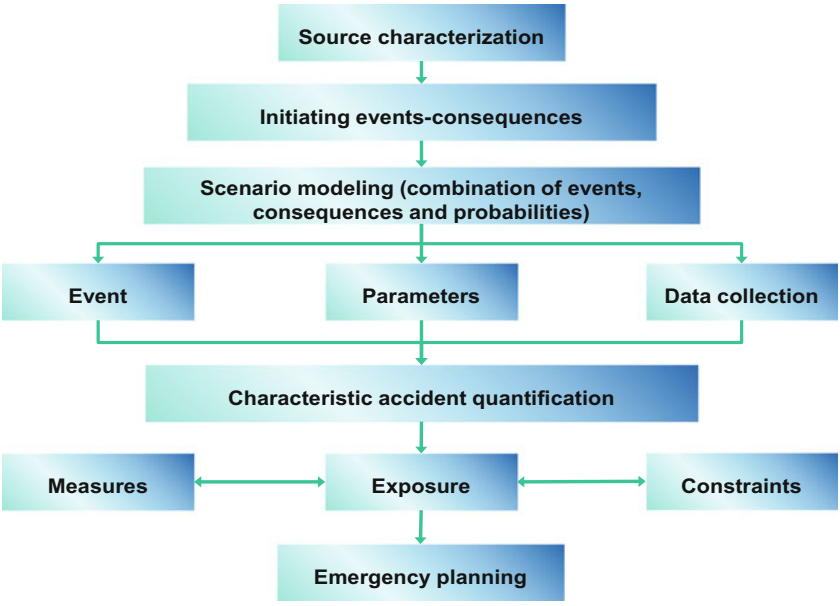


**Fig. 16.1** Potential exposure assessment procedure

operational processes that take place in it; and the deviations from normality which may lead to the loss of control of the radiation sources.

The availability of engineered safety systems—e.g., locking mechanisms; access control systems; physical barriers—and its features—diversity, redundancy and segregation; failure independency; reliability—should be taken into account when assessing the initiating events and ways of potential exposures.

Let take for example the mentioned event of an industrial radiography source that fails to retract correctly; this could occur because:

- The source is stuck in the guide tube and… (this may be a device failure)
- The source is disconnected from the control cable and…(this may be a device failure, but also a failure to follow procedures)
- The radiographer did not fully retract the source and… (this may be a human error)
- The radiographer pushed the source out of its shielded position and… (this may be a human error)

Modeling scenarios may need the use of structured methods that combine initiating events and likely failures—e.g., of a safety system, device, software, or procedure—with consequences representing potential exposure conditions. These methods help to not overlook any possibility when gathering the lists of events. Known methods are event tree analysis, fault tree analysis, cause-consequence and event sequence diagrams [4, 5].

For example, it is possible to assume a realistic exposure from an entry into an irradiation room when the source is not in the shielding position; then, with a fault tree, identify the initiating events that could lead to such unnoticed entry. Furthermore, with an event tree, it is possible to quantify even more realistic outcomes from each failure and assign a probability to each node or branch point, based on operational experience or good judgment.

As shown in Fig. 16.2, event trees usually start with an initiating demand—jump from the airplane in the example—and move through successive responses, describing the outcome in terms of success or failure of individual steps and/or devices (main and reserve chutes fail or work). Fault trees begin at the other end, with a specified unwanted outcome—the reserve chute fail in the example—and work backward to analyze possible ways in which this outcome could have occurred.

Just to mention some assumptions, the scenario modeling for the previous stuck radiographic source example may include causes and consequences like these:

- No survey was performed; the source was unknowingly kept exposed—out of its shielding—for several hours while members of the public were working around… (potential public exposure at levels depending on time and distance)
- The assistant manipulated the tube where the source was unnoticed without surveying it, after which the source fell to the ground… (possible exposure of hands and other body regions depending on the position, on how many times he touched the tube, and the time he stayed near the source)

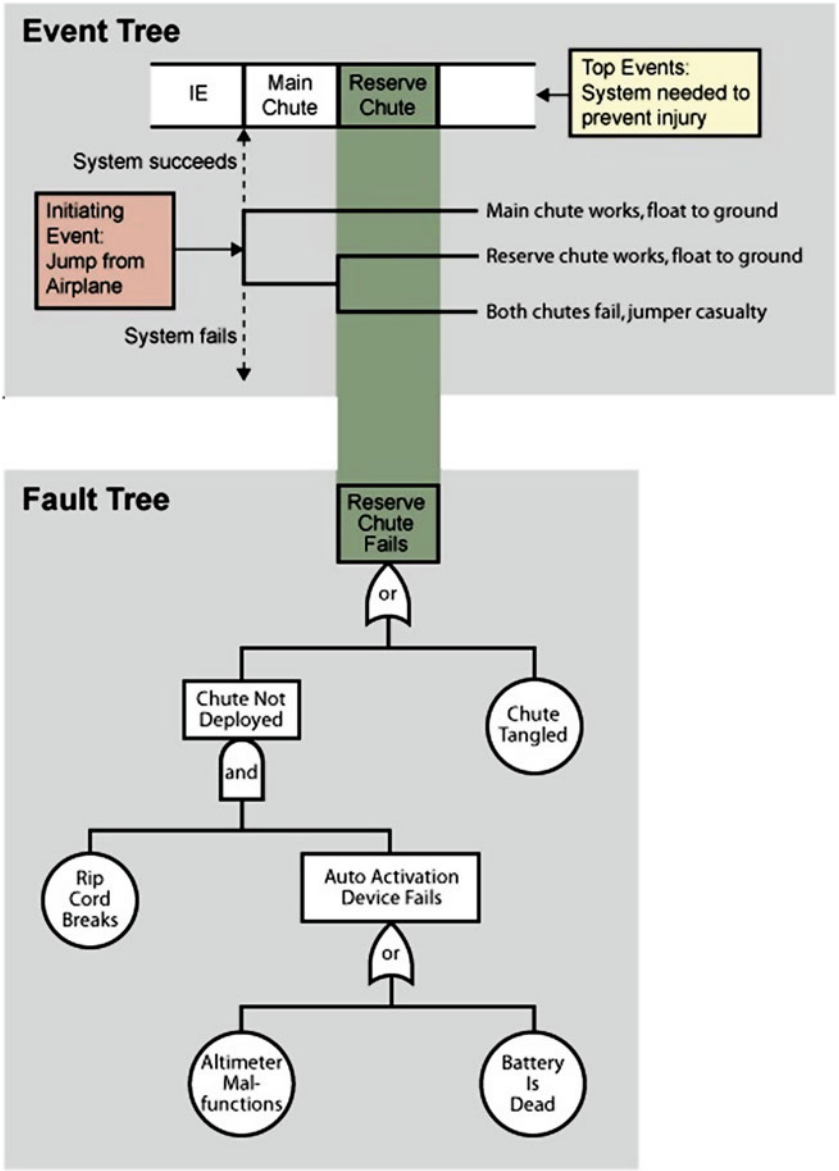


**Fig. 16.2** Sample of PRA (Courtesy of U.S. Nuclear Regulatory Commission)

- Somebody picked up the dropped source and put it in one of his trouser pockets, where the source remained for several hours… (potential exposure to the hands depending on how many times he handled the source, severe potential exposure

to the inferior extremities and pelvic region…, likely amputation of various body parts)
- The individual who picked up the source took it home… (potential exposure of family members: spouse, children, the severity depending on time and distance)

Unfortunately these events have happened in real life. The main victims of accidents involving industrial radiography sources have been members of the public and other workers not related to the source, with local irradiations above the threshold for deterministic effects that have required amputation of fingers, hands, limbs, and several surgical interventions with skin transplanting (skin grafts), and even death [6].

Another important task in potential exposure analysis is the accident-sequence quantification, for which data collection—e.g., failure frequencies, system inadequacies, human errors—and parameter definition are needed. Despite the large uncertainties in estimating the probability of an unsafe situation, and the resulting doses and risks, case specific risk constraints can be used along with dose constraints as criteria for the analysis.

Final results are relevant to determine the measures to prevent events to happen or reduce their likelihood. Depending on the source or facility under analysis, recommended measures could be as simple as reinforcing existing control procedures and personnel training, or installing new and more reliable safety blocking and alarming devices, or safety systems.

## 16.3 Accidents and Consequences

From 1945 to 2007, a total of 203 accidents involving radiation sources, excluding malicious acts and nuclear testing, were reported by the UNSCEAR [7]. Some of these accidents have resulted in significant health effects and, occasionally, in fatal outcomes.

UNSCEAR data of radiological accidents from this period that resulted in acute health effects or significant public exposure is shown on Fig. 16.3 by application. In Fig. 16.4, the same data is arranged by period. According to UNSCEAR [7], almost 74 % of the 19 accidents at nuclear facilities—critically and non-critically with on-site consequences and release to the environment—were associated with nuclear weapons programs, and occurred on the early period after the World War II and 1965. The period for development and introduction of power reactors in industry—1966–1986—accounts for 12 accidents, mainly in research facilities and zero power reactors.

In contrast, commercial nuclear facilities, with more than 400 power reactors in operation, have reported, until now, only three major accidents with releases to the environment. These are:

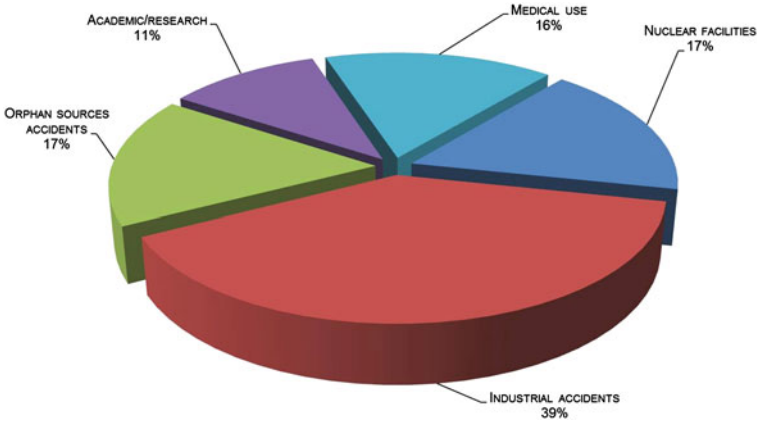


**Fig. 16.3** Accident data reported by the UNSCEAR for the period 1945–2007 by application

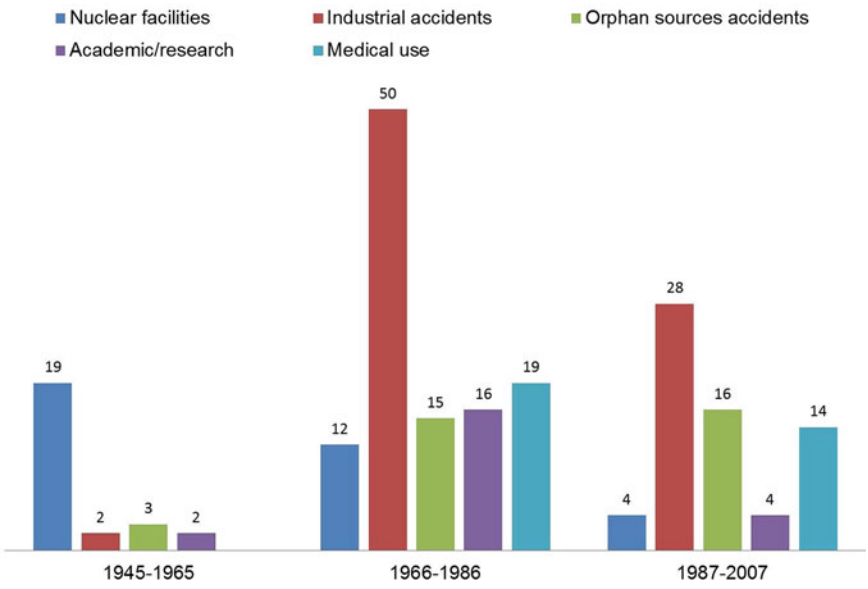


**Fig. 16.4** Same data reported by the UNSCEAR arranged by periods

1. The Three Mile Island nuclear power plant accident in 1979, where a sequence of events led to core melt and the release of fission products in the primary system. Although large amounts of $^{133}$Xe and $^{131}$I were released into the atmosphere, the resulting exposures to the public were negligible: just about 0.01 mSv [8];
2. The Chernobyl nuclear power plant accident in 1989, where 28 individuals from the plant staff and emergency personnel died from severe radiation effect while

responding to the steam explosion that destroyed the Unit 4. Over 200,000 people were evacuated and resettled, and a significant amount of $^{131}$I and $^{137}$Cs was released into the environment and contaminated large areas of Belarus, Russia, and Ukraine [9], but without exposing the general population to harmful radiation doses nonetheless;

3. The Fukushima Daiichi nuclear power accident in 2011, which led to severe core damage to three of the six nuclear reactors on the site, and the release, over a prolonged period, of large amounts of $^{131}$I, $^{137}$Cs, and $^{134}$Cs that were dispersed over the North Pacific Ocean. A significant amount of $^{131}$I and $^{137}$Cs was released (100–400 PBq and 7–20 PBq, respectively) and more than 100,000 people were evacuated. The estimated dose received by the adult population was, on average, less than 10 mSv. From the group of on-site emergency workers, only 174 (around 0.7 %) received doses in excess of 100 mSv, six of whom exceeded the temporary dose criterion of 250 mSv. Neither severe deterministic effect nor deaths were reported [10, 11].

From these accidents it has been also learned that social and psychological consequences from an undue fear of radiation can be worse than the direct radiological impact.

The decrease in accidents at nuclear facilities over the years is obvious from Fig. 16.4. In contrast, industrial and orphan source accidents have increased and are now accountable for more than 50 % of total accidents. Accidents associated with industrial radiographic sources are grouped within industrial accidents. Only a few accidents have been reported regarding the use of accelerators, research reactors, radiochemistry laboratories, small radiation facilities and the use of *X*-ray units for different analysis.

The numbers of deaths and early acute health effects due to radiological accidents are itemized in Fig. 16.5. Although these data may not be totally completed and updated, it is a fact that the total number of deaths (147) from radiological accidents over a period of more than 60 years, including those from the Chernobyl

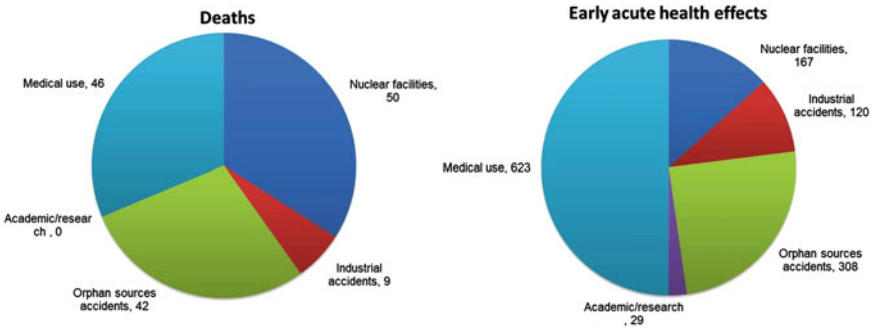


**Fig. 16.5** Numbers of deaths and early acute health effects from radiological accidents for period 1945–2007 by application

accident, is by far much lower than the tens of thousands of deaths annually caused from slips, trips, and accidental electrocutions in the common industry.

The higher number of people who experienced early acute health effects in medical use of radiation is also evident from Fig. 16.5; but, it is important nonetheless to bear in mind the extremely large number of medical procedures performed annually all over the world.

Some nonnuclear accidents with fatal victims from the UNSCEAR report [7] are listed apart in Table 16.1 to illustrate the data shown in Fig. 16.5. Family members, including children, other members of the public, and patients are indicated in the "Other" column. Casualty incidences of accidents associated with orphan sources—spent and disused sources not properly disposed—can be clearly noticed.

**Table 16.1** Reported accidents, other than NPP accidents, with fatal victims during period 1945–2007

| Year | Location | Cause | Fatalities | |
|---|---|---|---|---|
| | | | Workers | Other |
| 1962 | Mexico: Mexico City | Abandoned $^{60}$Co source | | 4 |
| 1963 | China: Hefei City | Abandoned $^{60}$Co source | | 2 |
| 1966 | Russia: Kaluga | *X*-Ray patient overexposure | | 1 |
| 1968 | USA: Wisconsin | $^{198}$Au higher dose administered | | 1 |
| 1975 | Italy: Brescia | $^{60}$Co irradiation facility improper entry | 1 | |
| 1975 | Ukraine: Sverdlovsk | $^{60}$Co source dropped during transport | 1 | |
| 1978 | Algeria | $^{192}$Ir source fell during transport, then picked up by somebody | | 1 |
| 1980 | Russia: Leningrad | $^{60}$Co irradiation facility improper entry | 1 | |
| 1980 | Russia: Yuzhno-Sakhalinsk | $^{192}$Ir source improperly stored, picked up by two children | | 1 |
| 1982 | Norway: Kjeller | $^{60}$Co irradiation facility improper entry | 1 | |
| 1982 | Azerbaijan | Abandoned $^{137}$Cs sources | | 5 |
| 1984 | Morocco | Dropped $^{192}$Ir source picked up and taken home by somebody | | 8 |
| 1985 | China: Mudanjiang | Abandoned $^{137}$Cs source | | 1 |
| 1986 | USA: Texas | Patient accelerator overexposure | | 2 |
| 1987 | Brazil: Goiania | Abandoned $^{137}$Cs source | | 4 |
| 1989 | El Salvador: San Salvador | $^{60}$Co irradiation facility improper entry | 1 | |
| 1990 | Israel: Soreq | $^{60}$Co irradiation facility improper entry | 1 | |
| 1990 | China; Shanghai | $^{60}$Co irradiation facility improper maintenance | 2 | |
| 1990 | Spain: Zaragoza | Patient accelerator overexposure | | 15 |

(continued)

**Table 16.1** (continued)

| Year | Location | Cause | Fatalities | |
|---|---|---|---|---|
| | | | Workers | Other |
| 1991 | Belarus: Nesvizh | $^{60}$Co irradiation facility improper entry | 1 | |
| 1991 | United Kingdom | Industrial radiography chronic exposure due to improper operation | 1 | |
| 1991 | Ukraine | $^{137}$Cs source found embedded in bedroom wall | | 2 |
| 1992 | China: Xinzhou | Abandoned $^{60}$Co source | | 3 |
| 1992 | USA: Pennsylvania | Brachytherapy source remained in patient | | 1 |
| 1994 | Estonia: Tammiku | Stolen $^{192}$Ir source | | 1 |
| 1995 | Russian Federation | Abandoned $^{137}$Cs source in a truck for $\sim$5 months | | 1 |
| 1996 | Costa Rica: San Jose | $^{60}$Co patient overexposure | | 17 |
| 2000 | Thailand: Samut Prakan | Abandoned $^{60}$Co source | | 3 |
| 2000 | Egypt: Meet Halfa | Dropped $^{192}$Ir sources found by a farmer | | 2 |
| 2001 | Panama: Panama City | $^{60}$Co patient overexposure | | 5 |
| 2004 | France: Epinal | Patient overexposure | | 4 |

Most frequent causes of fatalities and severe injuries have been improper entries into the irradiation room (China 1972; Moscow 1973; New Jersey 1974; El Salvador 1989; Israel 1990; etc.); mishandling of industrial radiography sources (Morocco 1984; Ukraine 1991; Peru 1999; Egypt 2000; etc.); disassembling abandoned teletherapy sources (Mexico 1983; Goiania 1987; Turkey 1993; etc.); and overexposing patients during treatment (Spain 1990; Costa Rica 1996; France 2004; etc.).

Two typical examples: the 1987 accident in Goiania killed 4 people, injured 28 and produced 3,000 m$^3$ of waste contaminated with $^{137}$Cs. The 1983 accident in Ciudad Juarez, Mexico, generated a large scale contamination; 21 areas contaminated with $^{60}$Co were identified, 109 houses built with contaminated rebar were demolished, and 37,000,000 kg of contaminated rods, metallic bases, material in process, scrap metal, etc., were produced.

## 16.4 The International Nuclear and Radiological Event Scale (INES)

The International Nuclear and Radiological Event Scale (INES), co-sponsored by the IAEA and the Nuclear Energy Agency (OECD/NEA), was introduced in 1990; it is aimed at facilitating communication and understanding between the technical

community, the media and the public, on the safety significance of the different events that may occur [12].

INES covers a wide spectrum of practices, including industrial radiography, the use of radiation sources in medicine, operations at nuclear facilities, and the transport of radioactive material; however, in medical applications, INES can be used for the rating of events resulting in actual exposure of workers and the public, but not for the rating of actual or potential consequences for patients exposed as part of a medical procedure.

With INES, radiological and nuclear accidents can be categorized in different levels depending on the release of radioactive material and the extension of areas affected by contamination. For incidents with a lower impact on the environment, the rating is based on the doses assessed and the number of people exposed. As shown in Fig. 16.6, events are classified at seven levels. Levels 4–7 (above the dash line) are called "accidents", while levels 1–3 (below the dash line) are named "incidents". Events without safety significance are known as "Below scale/Level 0".

For communications purposes, levels are identified in order of increasing severity as: anomaly; incident; serious incident; accident with local consequences; accident with wider consequences; serious accident; and major accident. In accordance with INES, Chernobyl and Fukushima accidents were rated as Level 7 (major accidents), while the Three Mile Island accident and the Goiania accident were rated as Level 5 (accidents with wider consequences) [13].

INES does not replace the existing reporting requirements for notifying the NRC of emergency and non-emergency events. They include reports of theft or loss of licensed material, reports of releases and exposures, notifications of failures, report and notification of a medical event, etc. [14].

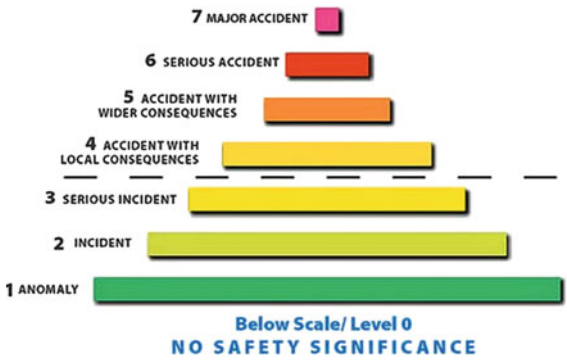


**Fig. 16.6** Nuclear and Radiological Event Scale (INES) (Courtesy of U.S. Nuclear regulatory Commission)

## 16.5 Emergency Planning and Preparedness

Emergency plans to respond if an exposure or contamination occurs are required anywhere radiation sources are used. Emergency plans are developed at the activity or facility, according to the magnitude and probability of exposures that may cause the highest effects (high risk/low frequency) or have the highest probabilities (low risk/high frequency), since the last suggests safety weaknesses and may lead to successive important failures.

The emergency plan at the facility usually contains two main parts; the first is an overview of the emergency situations predicted from the safety assessment, and describes the procedures to avoid severe deterministic effects; the accident overview also includes parameters and indicators to detect deviations from normality, and communication deadlines. The second part provides a detailed account of the roles and responsibilities; the specific tasks to be performed to respond to the emergency; the resources assigned—communication and alerting devices, measuring instruments, personal protective equipment (PPE), safety signs and labels, source recovery tools, additional shielding materials, decontamination materials, first aid kits, empty containers for radioactive waste, etc.—and where they are located. These resources should be available, identified and in good standing at all times.

Under role and responsibilities, the plan undoubtedly appoints the individuals with the adequate authority to classify any incident and, upon classification, promptly, and without consultation, to initiate the appropriate response, and notify the off-site officials and first responders per plan provisions. Under tasks, the plan will include all reasonable actions to protect workers and members of the public from exposure; the means to protect emergency personnel during response operations, and to gather all useful information to evaluate possible health effects and prevent similar situations in the future.

The arrangements made and documented to obtain additional assistance in case of emergency should also be part of the plan; for example, support agreements from local damage control, firefighting, and radiological emergency teams. The sort and extent of these arrangements are suitable for the potential threat associated with the facility or activity.

## 16.6 National Response Framework

The radiological emergency plan at facility level is also interrelated with the operational planning for disaster and major incidents at local, tribal, state, and national levels [15]. There are special teams prepared to implement protective actions at these levels, whether the risk comes from a power plant, fuel cycle facility, radiopharmaceutical manufacturer or user, or waste management facility, or from a lost or abandoned radioactive source.

The National Response Framework is how the nation responds to all types of disasters and emergencies. The preparedness and immediate response resources at federal and local levels in case of emergencies involving exposure to ionizing radiation or radioactive material releases within this framework are outlined in Fig. 16.7.

The federal response to a specific incident is based on several factors, including the ability of state, tribal, and local officials to first respond; the type, amount, and authority over the radioactive material involved; the scale of the potential impact on the public and environment; and the extent of the affected area [16].

The Federal Radiological Preparedness Coordinating Committee (FRPCC) is the national-level setting for the development and coordination of radiological planning and preparedness policies and procedures. The FRPCC is an interagency body, chaired by the Department of Homeland Security and the Federal Emergency Management Agency (DHS/FEMA), that brings together the Coordinating agencies —Environmental Protection Agency (EPA), Department of Energy (DOE), Department of Defense (DOD), and Nuclear Regulatory Commission (NRC)—with other Cooperating Federal agencies that provide additional technical and resource support as the Department of Health and Human Services (HHS), including the

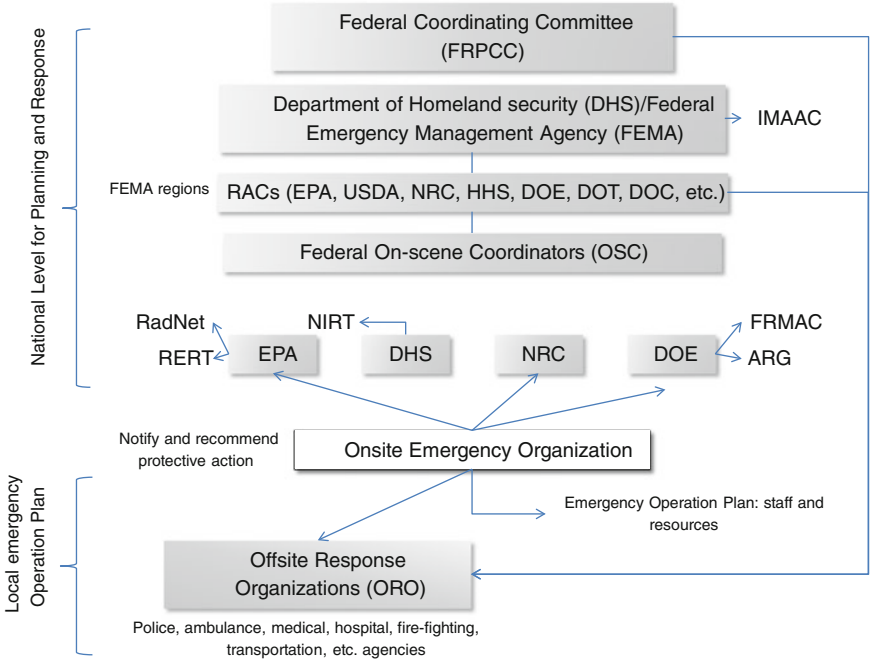


**Fig. 16.7** National response framework

Centers for Disease Control and Prevention (CDC); the Food and Drug Administration (FDA); the Department of Agriculture (USDA); the Department of Commerce (DOC); the Department of Transportation (DOT), etc. [17].

In FEMA regions, the primary coordinating structures at Federal Regional level are the Regional Assistance Committees (RACs). RAC membership mirrors that of the FRPCC.

Figure 16.7 also shows some of the most important specialized resources for radiological response. They are:

- The Interagency Modeling and Atmospheric Assessment Center (IMAAC), which provides the single Federal atmospheric prediction of hazardous material concentration to all levels of the Incident Command;
- The Federal Radiological Monitoring and Assessment Center (FRMAC), which is available on request to respond to nuclear/radiological incidents; this center is usually located at an airport near the scene of the radiological emergency, and is responsible for coordinating all environmental radiological monitoring, sampling, and assessment activities for the response and site cleanup. DOE leads the FRMAC for the initial response, then transitions FRMAC leadership to EPA for site cleanup;
- The DOE Accident Response Group (ARG), which includes scientists, technical specialists, crisis managers, and equipment ready to respond to a nuclear weapon accident;
- The Nuclear Incident Response Team (NIRT) consisting of the DOE resources and EPA entities under DHS direction; the team is activated when DHS, in consultation with EPA and DOE, determines that the severity of the incident merits the NIRT assets to perform radiological emergency response functions;
- The EPA Radiological Emergency Response Team (RERT), which provides resources, including personnel, specialized equipment, technical expertise, and laboratory services to aid coordinating and cooperating agencies and state, tribal, and local response organizations in protecting the public and the environment;
- The RadNet, an EPA system of fixed and deployable monitoring stations to provide for a nationwide environmental monitoring network for radiological impact assessment.

Federal and state capabilities are to respond to a wide range of situations, including:

- Inadvertent or accidental incidents in fixed nuclear facilities;
- Deliberate attacks involving nuclear or radioactive materials or radiation sources;
- Lost/found/orphaned radioactive material sources;
- Transportation incidents;
- Domestic nuclear weapons accidents; and
- Foreign incidents involving nuclear or radioactive material.

The EPA is the coordinating agency for the federal environmental response to incidents that occur at facilities not licensed, owned, or operated by NRC or an Agreement State, or currently or formerly licensed facilities for which the owner/operator is not financially viable or is otherwise unable to respond [16].

In regard to radioactive sources that pose a potential risk to health, safety, and national security, DOE had removed more than 35,700 excess, unwanted, abandoned, or orphan radioactive sealed sources through the Off-Site Source Recovery Project (OSRP) managed at Los Alamos National Laboratory [18], in an effort to reduce this threat.

## 16.7 EPA Protective Action Guides

Actions to protect the public, such as sheltering and evacuation, are generally recommended when a relatively significant release of radionuclides is possible and the radiological risk is weighed against other non-radiological detrimental consequences, such as economic, social, and psychological. Yet, when there is no time for a deep analysis, decisions must be taken immediately based on previously accepted criteria.

Previously accepted criteria are the EPA Protective Action Guides (PAGs); that is, the projected doses that would trigger measures, like sheltering and evacuation of the public in the closeness of the accident site. PAGs are implemented by federal, state, tribal and local authorities depending on the projected dose to an individual in the general population. Depending on the accident, the closeness is determined by the plume exposure pathway, or the damage and/or fallout zone [17].

Some relevant projected doses associated to recommended protective actions are shown in Table 16.2 [15]. These actions are planned to be conducted in an early phase of the accident to reduce the direct exposure from the passing cloud, and from then on, the exposure from deposited radioactive materials.

**Table 16.2** EPA protective action guides

| Protective action recommendation | Protective action guide (PAG) |
|---|---|
| Sheltering-in-place or evacuation of the public | 10 mSv to 50 mSv projected dose over 4 days |
| Administration of prophylactic drugs—KI | 50 mSv projected child thyroid dose from radioactive iodine |
| Relocation of the public | 20 mSv projected dose in the first year, subsequent years, 5 mSv/year projected dose |
| Food interdiction | 5 mSv/year projected dose, or 50 mSv/year to any individual organ or tissue, whichever is limiting |
| Apply simple dose reduction techniques | 20 mSv |

Concerning the protection of workers, all efforts have to be made not to expose any individual in excess of the dose limit for occupational exposure of 50 mSv in a single year. 100 mSv is the worker guideline for protecting valuable property necessary for public welfare, and 250 mSv, the worker guideline for lifesaving or protection of large populations.

## 16.8 Emergency Training

Training is an essential part of the emergency preparedness; hence, it is recommended to periodically conduct full-scale and functional exercises, drills, tabletop exercises, seminars, workshops and courses, on potential emergency situations, designed for building capability and improve the emergency response.

All facility personnel should receive emergency training at least once a year on the organization-specific procedures. Topics such as dose measurement, communication, waste disposal and decontamination procedures, the use of personal protective equipment (PPE), collection and analysis of samples, first aids, and the like, can be covered by training games and drills, alone or combined. Potential emergency situations, deviation indicators, and planned procedures can be covered by seminars and tabletop exercises.

Specific training should also be provided on a regular basis to:

- Incident commanders
- First responders,
- Radiological monitoring and radiological analytical teams,
- Clinicians, health practitioners, and hospital staff,
- Police, security forces and firefighters,
- Journalists,
- Response volunteer organizations such as the American Red Cross and the Medical Reserve Corps, and
- Decision makers at all levels.

Full-scale and functional exercises are simulations for major scale accidents, which are used to allow different groups and organizations to act and interact in coordination. Functional exercises are performed to evaluate specific functions, e.g., coordination, command and control, between various multi-agency coordination centers. In functional exercises, personnel usually perform specific duties in a simulated operational environment.

Full-scale exercises are as close to the real thing as possible. They typically simulate an emergency situation at a facility with off-site radiological release, and require a real response by all response organizations, including:

- Licensee, local, state and federal responders;
- Incident management and multi-agency coordination centers;
- Senior decision makers;
- Recovery stakeholders.

NRC requires full-scale exercises at nuclear plants to be performed at least once every 2 years. Additional drills are conducted in-between these 2 year exercises. These exercises are evaluated by NRC inspectors and FEMA evaluators. For example, a full-scale exercise was conducted in South Carolina, in the past year, with the participation of the IAEA and Canada, to demonstrate the effective information sharing among international, federal, state, and local responders during the response to a nuclear power plant accident [19].

Here are some examples of scenarios, other than releases from nuclear power plants, which possibly require an off-site exercise response [20]:

- The incineration of a $^{60}$Co source in a smelter;
- A fire in a LILW storage facility;
- A fire in a radiopharmaceutical storage facility;
- A radiation emergency involving transport, missing source, and overexposure;
- A package exhibiting high radiation readings that arrives at an airport;
- A radioactive material found in an apartment block; etc.

The FEMA Radiological Emergency Preparedness Program (REP) coordinates the National effort to provide state, local, and tribal governments with relevant training and exercise guidance to ensure that adequate capabilities exist to prevent, and respond to, any radiological accident [21].

## References

1. International Atomic Energy (2014) Radiation protection and safety of radiation sources: international basic safety standards. IAEA safety standards series GSR Part 3, IAEA, Vienna
2. Kosow H, Gaßner R (2008) Methods of future and scenario analysis. Overview, assessment, and selection criteria. Deutsches Institut für Entwicklungspolitik gGmbH, Bonn
3. ICRP (2007) The 2007 recommendations of the international commission on radiological protection. Ann ICRP 37(2–4)
4. International Atomic Energy Agency (2002) Procedures for conducting probabilistic safety assessment for non-reactor nuclear facilities. TECDOC-1267. IAEA, Vienna
5. Vesely WE, Goldberg FF, Roberts NH, Haasl DF (1981) Fault tree handbook. NUREG-0492. U.S. Nuclear Regulatory Commission, Washington
6. Ortiz P, Oresegun M, Wheatley J (2000) Lessons from major radiation accidents. In: *IRPA-10.* International congress of the international radiation protection association, Hiroshima
7. UNSCEAR 2008 (2011) Sources and effects of ionizing radiation. Report to the general assembly with scientific annexes volume II. Scientific Annex C. radiation exposure in accidents. UN, New York
8. U.S. NRC (2014) Backgrounder on the Three Mile Island Accident," United States Nuclear Regulatory Commission. http://www.nrc.gov/reading-rm/doc-collections/fact-sheets/3mile-isle.html
9. U.S. NRC (2014) Backgrounder on Chernobyl nuclear power plant accident. United States Nuclear Regulatory Commision. http://www.nrc.gov/reading-rm/doc-collections/fact-sheets/chernobyl-bg.html
10. International Atomic Energy Agency (2015) The Fukushima Daiichi accident. Report by the director general. IAEA STI/PUB/1710. IAEA, Vienna

11. UNSCEAR (2014) Source, effects and risks of ionizing radiacion. Report to the general assembly with scientific annexes volume I. Annex A. Levels and effects of radiation exposure due to the nuclear accident after the 2011 great east-Japan earthquake and tsunami. UN, New York
12. International Atomic Energy Agency & Nuclear Energy Agency (2008) INES The international nuclear and radiological event scale 08-26941/E. IAEA, Vienna
13. International Atomic Energy Agency and OECD/Nuclear Energy Agency (2013) The international nuclear and radiological event scale user's manual 2008 edition, INES. IAEA, Vienna
14. U.S. NRC (2014) Reporting requirements. United States Nuclear Regulatory Commission. http://www.nrc.gov/about-nrc/emerg-preparedness/faq/reporting-requirements.html
15. U.S. Environmental Protection Agency (2013) PAG manual. Protective action guides and planning guidance for radiological incidents
16. DHS (2008) Nuclear/Radiological incident annex to the national response framework. DHS
17. DHS/FEMA (2015) The nuclear/radiological incident annex to the national response framework. DHS/FEMA, Washington
18. Los Alamos National Laboratory (2015) Off-site source recovery project, OSRP. U.S. Department of Energy's National Nuclear Security Administration. http://osrp.lanl.gov/
19. NRC, FEMA and DOE/NNSA (2015) 2015 Full-scale nuclear power plant exercise. NRC, FEMA, and DOE/NNSA
20. International Atomic Energy Agency (2005) Preparation, conduct and evaluation of exercises to test preparedness for a nuclear or radiological emergency. IAEA, Vienna
21. FEMA (2016) Radiological emergency preparedness program. Department of Homeland Security. https://www.fema.gov/radiological-emergency-preparedness-program

# Chapter 17
# Radiation Protection Program Details

As provided by 10 CFR § 20.1101, each licensee shall develop, document, and implement a radiation protection program commensurate with the scope and extent of licensed activities, and sufficient to ensure compliance with the regulations [1]. The radiation protection program details for various applications that could be of interest, and used as an information source for other programs, are discussed below.

Based on information gathered from previous chapters, a radiation protection program should ensure that:

1. Responsibilities and authorities for safety are established;
2. Requirements to control the sources and protect the workers—radiation shielding, containment, access control, ventilation, contamination control, surveillance and monitoring, etc.—are met;
3. Procedures governing the use and control of radiation sources are settled, documented, and followed in conformity with regulations and licensing requirements, and
4. Appropriate assessments and reviews are properly scheduled and documented to confirm the achieved worker's and public's safety.

In brief, the radiation protection program should answer to all questions about how, what, when, where, how often, and by whom, something has to be done to ensure safety.

## 17.1 Irradiation Facilities

Irradiation facilities use high intensity gamma sources, accelerators, or *X*-ray generators, to deliberately and safely irradiate materials, such as food, blood, health care products, polymers, etc., for commercial or research purposes. $^{137}$Cs sources, with activities of several tens to several hundreds of TBq and *X*-ray generators, are used for blood irradiation, and $^{60}$Co sources, with activities that can reach hundreds



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of PBq, are used for food irradiation and health care product sterilization. Accelerators with electron beams of 5−10 MeV are also used for sterilization of medical supplies and pharmaceuticals, since there is no induced radioactivity in any part of the equipment at these energies.

On the basis of the design and, in particular, the accessibility and shielding of the radioactive source, there are four types of gamma irradiators: self-shielded irradiators; panoramic irradiators with dry source storage; underwater irradiators; and panoramic irradiators with wet source storage. Irradiators using electron beams and *X*-rays are typically integrally shielded units or are constructed with the unit housed in a shielded room.

Irradiation facilities must be authorized by a NRC's license for the use of licensed material. Before the irradiation facility is constructed and operated, a construction license is also required; this means that these facilities should undergo construction monitoring and acceptance testing before being licensed. Licenses requirements for commercial irradiators are stated in 10 CFR Part 36 [2], while licenses requirements for self-shielded irradiators are specified in the guidance NUREG 1556 Vol. 5 [3]. Licenses requirements for accelerators can be found in the Suggested State Regulations for Control of Radiation [4].

The radiation protection program should have an initial statement outlining the safety policy and accountability for safety, including the responsibilities and authorities of all departments, groups or individuals, using gamma irradiators or accelerators for commercial, research or development purposes. Such statement may appear like a paragraph indicating the safety priorities and the purposes the program is designed for. It is also necessary to identify the source locations at the facility and describe their technical specifications.

It is also appropriate to refer to each source license (or registration) number and docket, and the main regulations that serve as basis for the radiation protection program. Besides, source locations—e.g., irradiation room, source storage, waiting room for spent sources, etc.—should be drawn in a map indicating the boundaries of controlled and supervised areas around the sources, and the established points for workplace monitoring.

Irradiators should be located and constructed so that the exposure level in any unrestricted area (e.g., an office or the exterior surface of an outside wall) does not exceed 1 mSv a year, and the dose rate does not exceed 20 $\mu Sv\ h^{-1}$ [2]. For panoramic irradiators, the dose rate is measured with the radiation sources exposed, at a distance of 30 cm from the room wall. For dry storage irradiators, the dose rate is also measured at a distance of 1 m from the shield with the source shielded. For a pool irradiator, the 20 $\mu Sv\ h^{-1}$ dose rate at 30 cm is measured over the edge of the pool.

Controlled rooms are typically the rooms in which radiation sources are housed and exposed, and may include the roof for panoramic irradiators. Supervised areas usually include control rooms, product entry and exit areas, and service areas; for underwater or wet storage gamma irradiators, the water treatment rooms are also classified as supervised areas; and power supply rooms, for electron beam irradiators.

A radiation safety officer (RSO) should be appointed at the operational organization; the organizational structure responsible for irradiation, specifically, all management personnel who have important radiation safety responsibilities or authorities, including the radiation safety officer, should be as well clearly stated and documented as part of the radiation protection program [2].

Underwater and wet storage irradiators use a water pool as gamma shielding. Irradiation rooms for panoramic irradiators are typically labyrinths constructed with thick concrete walls ($\approx$ 2 m) to maintain the dose rate in surrounding areas close to background levels. Shielding maze configuration allows for the movement of the product, while significantly reduces scattered radiation reaching operators. Product containers that are going to be irradiated are either continually moved around the unshielded radiation source—using a conveyor or hanging from a track on the ceiling—or are arranged in the irradiation room before exposing the source. Shield penetrations for personnel and product entry and exit ports, ventilation system, and other service ducting, are usually designed with shield plugs made of lead or steel shot to prevent radiation leakage; tubes, pipes, and conduits are installed in a curved or stepped pathway through the shielding material, so as to ensure that there is no direct radiation leakage. The shield and maze design for electron beams operating above 10 MeV, and *X*-ray irradiators operating above 5 MeV, should also take into account the occurrence of neutron radiation fields and activation products.

Irradiators require a forced ventilation system to remove the ozone and other toxic gases that can be formed due to air radiolysis[1]; the ventilation system should maintain a negative pressure in the irradiator to prevent the migration of ozone to occupied areas.

Engineering controls are to be installed to automatically restore the source to its shielded position, or automatically switch off the radiation beam of accelerators and *X*-ray irradiators, at any attempt to gain access to the controlled area when the irradiation is taking place. Access control interlocks should be typically redundant and independent, that is, if one system fails, there is yet another system based on a different principle as a backup; access interlocks include pressure mats; light beam interruption detectors (photo eyes); gates or doors that open only to allow product to pass through and then close immediately; detectors that require that a product carrier always be present in the opening; multipurpose key—used both to operate the control console and to gain access to the radiation room—attached to a portable survey instrument, or a captive key (one key remains captive while the other is in use); and continuous radiation monitors; among others, that not only will interrupt the irradiation when the control mechanism has malfunctioned or been overridden or tampered with, but also actuate visible and audible alarms [5, 6].

Other controls dependent on the irradiator type include pool water temperature detectors, pool water level sensors, source rack position indicators, source status indicators, source guards and collision detectors, cooling water monitors, ozone and dark current time delay mechanisms, travel timers, intrusion alarms, emergency

[1]Radiolysis is the dissociation of molecules by ionizing radiation.

stops, ventilation system flow sensors, earthquake sensors, etc. All equipment inside the irradiation room, including wiring, electrical equipment, notices and lighting, should be selected so as to minimize failure due to prolonged exposure to radiation [5, 6]. Warning symbols and signs should be used in conjunction with visible and audible alarms.

Procedures and local rules are required for operation, maintenance, leak testing, water quality control, individual and workplace monitoring, fixed and portable instrument testing, interlock checking, and safety and warning devices checking, etc. In addition to product loading and unloading, control console operation, and other topics inherent to the irradiation process, relevant operating procedures should include instructions for entering the irradiation room after a delay—for ozone or spurious radiation—securing the room before leaving, responding to visible and audible alarms, using the emergency stop, and so forth. Procedures should also include gamma source loading and unloading operations, its reception, the storage, shipping and transportation of spent gamma sources; the use of remote instruments to manipulate the sources in wet and dry storages, and, if applicable, the storage, transportation, and disposal of activated (contaminated) accelerator parts; along with procedures for supervision of compliance, incident reporting, and emergency planning (see more possible procedures in the Radiation Protection Program Development Checklist section).

The radiation protection program should also contemplate the frequency and content of the personnel training—initial, refresher, and on-the-job—including instruction to management and administrative staff. The general training should include the basic principles and measures in radiation protection, including units of measurement and measuring instruments; the irradiator operational and safety features; actions to be taken in an emergency; applicable regulations; local rules and procedures; etc. As established by regulations [2], the irradiator operator safety performance should be annually evaluated to ensure that regulations, license conditions, and operating and emergency procedures are followed.

Dose rate measurements at relevant locations should be performed: (1) before commissioning, to confirm the shielding suitability, (2) after a modification of the radiation room shielding structure or occupancy of adjacent areas, (3) following the introduction of new radioactive sources, or changes in the orientation, energy or power level of the electron beam, and (4) on a routine basis at established points for workplace monitoring. Additionally neutron monitoring may be necessary if using electron beams with energies above 10 MeV or *X*-ray irradiators operating at 5 MeV or above. An adequate quantity of fixed and portable instruments, their annual calibration by an accredited organization, and a daily testing for proper operation should also be part of the program.

Although the risk of contamination is very low, irradiator sources should be regularly leak tested—e.g., every six months, and before placing them into a container or transport package—using a small standard laboratory wipe. For wet storage irradiators, it is recommended to install a fixed radiation monitor in the water treatment system to detect any contamination that may arise. Besides, water treatment filters and resin beds should be tested for contamination before its

removal, backwashing, or regeneration. If very-high energy electron beams or *X*-ray irradiators are used, periodic smear surveys of surfaces and air surveys should be scheduled to detect contamination from activation products. Source containers are checked for external and internal removable contamination at receipt, when returning the empty container, and at the time the spent sealed sources are sent back to the manufacturer.

Operators, radiation safety officers and maintenance staff normally entering controlled areas should be monitored using film badge dosimeters, thermoluminescent dosimeters (TLD), or optically stimulated luminescence (OSL) dosimeters. If the dosimetry system is provided by the irradiation facility, it should be traceable to a secondary standard laboratory; otherwise, the service should be arranged with an organization accredited by the Laboratory Accreditation Program (NVLAP). The program should also include the use of additional individual dosimeters with alarms.

Certainly, the radiation protection program should also consider its annual review. This annual review is to confirm that the program meets all regulatory requirements, is adequate to ensure the source, workers and public safety, and has been followed according to the expectations. The review should also serve to analyze the program faults and possible improvements.

Regarding recordkeeping, a copy of all licenses and its amendments, license conditions, and documents incorporated into the license by reference, should be kept until licenses are terminated. The documents supporting the facility and/or source designs—shielding, foundations, pool integrity, source rack, special form certification, etc.—should also be kept, as well as all records resulting from the radiation source receipt, transfer or disposal for as long as the sources are possessed, and for 3 years following the transfer or disposal [2]. Records of instrument calibrations, leakage tests, workplace monitoring, contamination surveys, inspection and maintenance, major malfunctions, significant defects, operating difficulties or irregularities, and major operating problems are kept for 3 years from the date of the test, calibration, measurement, or event. Training records should be retained until 3 years after the individual terminates work.

Since irradiator sources classify under IAEA source Category 1 [17], these facilities should have plans to handle a variety of emergencies, including malfunction or deliberate defeat of safety interlocks and access control systems; low water in, or water leakage from, the storage pool; leaking sources; automatic conveyor system jamming; source rack stuck in an unshielded position; prolonged loss of electrical power; fires; and expected natural phenomena (flood, earthquake, tornado, etc.); and the possibility of an accident during the transport of the radioactive sources [5]. Emergency procedures should include then actions, such as work interruption, irradiation shutdown and personnel evacuation, notification to the corresponding facility emergency levels, access restriction to specific areas, dose rate and contamination monitoring to determine further restriction areas, construction of a temporary shielding in the event that shielding integrity has been compromised, and speedy individual doses evaluation. The availability and location of emergency shielding and shielded containers, remote tools, and measuring

instruments should be part of the plan; also, the arrangements to receive external help, especially for people who can be injured or overexposed. The NRC requires irradiator operators to coordinate with local and state emergency response agencies in case of an emergency [2].

Irradiation facilities should be formally decommissioned. This means that the program should foresee, in case of license termination or permanent cessation of operations by come of age or other causes, the transfer of all gamma sources to a disposal operator or their return to the original manufacturer or distributor. For underwater and wet storage irradiators, it should also be anticipated the control of the pool water, its management and disposal as radioactive waste, and the decontamination of the infrastructure, if necessary. If accelerators have been used, its dismantling should be required, as well as, the management of large quantities of low active waste—i.e., generally less than 300 Bq/g for metal pieces of particle accelerators and less than 100 Bq/g for infrastructure materials such as concrete and reinforcement rods—from the activation of trace elements, and the use of recycle and reuse techniques to minimize the volume of radioactive waste going to a disposal operator. The decommissioning program should also include a comprehensive final monitoring to confirm the absence of contamination and any radiation source at the site, and the submission of all the corresponding documentation to the NRC or Agreement State.

## 17.2 Industrial Radiography

Industrial radiography is a nondestructive technique widely used in construction and industry for the inspection of welding assemblies, pipelines, pressurized piping, pressure vessels, storage containers, machined parts, etc. This technique uses a gamma source or *X*-ray generator and a film, or a radiation sensitive medium and detector, to produce a radiographic image, e.g., of a metal welding or aircraft part, to show its internal and surface defects, changes in structure, assembly details, etc. [8].

Industrial radiography poses little risk by itself; however, experience from past decades shows an alarming frequency of accidents with severe health consequences [9]. In view of this knowledge and the fact that radiographic testing is usually carried out under difficult conditions—in confined spaces, from scaffolding, in ditches, offshore platforms, pipe-laying ships, etc.—industrial radiography sources classify under IAEA source Category 2 [7].

Gamma industrial radiography should be authorized by a specific NRC's license for the use of licensed material; licenses requirements are stated in 10 CFR Part 34 [10]. *X*-ray machines for industrial radiography should be registered with the FDA, and its operation authorized by the state government [11]. As stated by regulations, to apply for a license or registration, the industrial radiography organization should perform a safety assessment for a range of scenarios representing normal use and

reasonably foreseeable incidents, including the potential exposures of radiographers, other workers and the public [10, 11].

The radiation protection program should have an initial statement outlining the safety policy and accountability for safety, including the responsibilities and authorities of all departments, groups, or individuals using gamma exposure devices, *X*-ray generators or accelerators for industrial radiography. Such statement may appear like a paragraph indicating the safety priorities and the purposes the program is designed for. It is also necessary to identify the source locations at the facility and describe their technical specifications.

*X*-ray generators and sealed gamma sources are commonly used in industrial radiography, both in fixed shielded enclosures—i.e., an enclosed shielded room, cell, or vault—with engineering controls; and in temporary jobsites using mobile or portable devices. All source locations—e.g., main source storage, workshop enclosed room, waiting room for spent sources, temporary storages at specific jobsites, etc.—should be in a map indicating the boundaries of controlled and supervised areas around the sources, and the established points for routine workplace monitoring.

A radiation safety officer (RSO) and a manager having authority regarding radiation safety should be appointed at the industrial radiography organization; also, a number of radiation protection officers should be designed to be available at client jobsites. All responsibilities and authorities should be clearly documented as part of the radiation protection program. In medium to large sized radiography organizations, a radiation safety committee may be recommended to regularly review the performance of the radiation protection program [12].

Conventional *X*-ray generators are used to perform panoramic and directional exposures. A panoramic exposure requires an annular beam to expose the full circumference of a cylindrical specimen. Collimators are necessary in directional exposures to restrict the beam size. *X*-ray sets with voltages up to 320 kV are generally useful for ambulatory work; while voltages up to 450 kV are better used in stationary or semi-ambulatory work. Betatrons and linear accelerators—portable, mobile—can be utilized for specific applications that require higher voltage *X*-rays [8].

Radiography devices containing sealed gamma sources are usually more portable than *X*-ray equipment, do not need power, and have better operability in temporary jobsites; they also are much higher energy than all but the most expensive *X*-ray equipment. Gamma exposure devices are classified according to their mobility as shown in Table 17.1. Table 17.1 also shows the ambient dose equivalent rate limits at 50 and 1 m from exposure device container established by ISO 3999:2004 [13].

Radiography devices can be shutter type if the gamma source stays in the device during the exposure, or projection type, if the source—in a flexible assembly or pigtail—is guided along a hollow tube out of the device to a collimator, with a

**Table 17.1** Maximum ambient dose equivalent rates in line with exposure device classification

| Mobility class | Description | At 1 m from external surface ($\mu Sv\ h^{-1}$) | At 50 mm from external surface ($\mu Sv\ h^{-1}$) |
|---|---|---|---|
| Class P | Portable exposure device, < 50 kg | 20 | 500 |
| Class M | Mobile, designed to be moved easily by a trolley or cart | 50 | 1000 |
| Class F | Fixed, installed or with restricted mobility in a shielded enclosure | 100 | 1000 |

control cable. A higher protection is provided by an S-bend projection type container, which holds the source near the center of a "dog-leg" or S-bend channel when in fully shielded position, and incorporates a flexible source holder, or pigtail, that can be secured at its cable-coupling end to the control cable port [12].

Gamma sources and *X*-ray tubes typically used for specific thicknesses are listed in Table 17.2. The table also shows some important gamma radionuclide features such as half-life and average energy.

The choice of equipment is based on various factors, the most important being the required energy to penetrate the object and detect the defect—thick materials require higher energy than thin materials—but also the object location, equipment maneuverability, and electric power accessibility.

*X*-ray equipment and gamma exposure devices should include all established relevant safety features and warning features to carry on industrial radiography works. For example, the *X*-ray generator control panel should be provided with a key switch, removable only in OFF or STANDBY position; fail-to-safe labeled warning lights indicating when the equipment is ready to emit *X*-rays, and when it is actually emitting; an ON switch requiring continuous pressure by the radiographer; a timer to control the exposure duration; and kV and mA indicators.

In projector type exposure devices, the locking mechanism designed to prevent unauthorized or accidental removal of the sealed source from its shielded position also has a key that should be removed and secured when the lock is fully engaged [10]. Besides, the shielding may incorporate depleted uranium (DU)—thicker than lead—for the safe storage of the gamma source, and to meet the requirements for

**Table 17.2** Gamma sources and *X*-ray used in industrial radiography examinations

| Steel thickness, mm | *X*-ray energy range | Gamma source | Average energy (MeV) | Half-life |
|---|---|---|---|---|
| 50–120 | 410 kV–4 MeV | $^{60}Co$ | 1.25 | 5.3 y |
| 12–70 | 275 kV–4 MeV | $^{192}Ir$ | 0.45 | 74 d |
| 8–30 | 175 kV–410 kV | $^{75}Se$ | 0.32 | 120 d |
| 4–20 | Up to 230 kV | $^{169}Yb$ | 0.2 | 32 d |
| 2.5–12.5 | Up to 130 kV | $^{170}Tm$ | 0.072 | 128 d |

Type B(U) packages under transport regulations [14, 15]. Thus, even the empty exposure device—without the radioactive source—should be safely stored and subject to accounting procedures.

Source changers—usually shipment containers as well—are shielded containers used to change the spent source by a new one. They should also have a lock, or outer locked container, to prevent the unauthorized or accidental removal of the sealed source, and should be kept locked at all times, except when under the direct surveillance of a radiographer or a radiographer's assistant [10].

Fixed *X*-ray or gamma source radiography machines are used inside shielded enclosures, preferably with a shielded roof, and equipped with functional door interlocks and audible/visual alarms [12]. A mechanical or electrical interlock system should ensure that the source cannot be exposed unless the door is closed. Maze room designs, as well as shield plugs, can be used to prevent or minimize shielding penetrations or scattering of radiation due to crane installations, pipework, control cables, ventilation ducting, etc. Emergency stop buttons or pull-cords with manual resets are also installed to enable any person within the shielded enclosure to terminate or prevent the radiation exposure.

To minimize the occupational radiation exposure at temporary jobsites, it is recommended the use of collimators—both for *X*-ray generators and gamma sources—and local shielding, such as lead sheets, bags of lead shot, and precast concrete structures.

Storage facilities are required to store gamma exposure devices and *X*-ray generators when are not in use. To avoid unauthorized accesses, gamma exposure device storage room should be designated as a controlled area, and *X*-ray generators storage room, as a supervised area. The doors to storage rooms are to be kept locked and the keys held by authorized personnel.

If required, arrangements can be made with the jobsite operator for the provision of suitable storage facilities for on-site overnight or between radiography sessions. A lockable room, purpose-built store, or storage pit, can do as an on-site storage; although, it should provide the same level of protection as storage facilities at the operating organization's main base.

Having and following procedures and local rules, is paramount for the safety of industrial radiography operations. Apart from regulating radiography operations in shielded enclosures and temporary jobsites, these procedures must cover routine device maintenance; on-site, and to and from temporary jobsites source transportation; gamma source exchange; exposure device withdrawals from storage; where, how, and when to measure radiation; source leaking testing; testing of radiation monitors; routine individual monitoring; and disposal or shipping of spent radioactive sources, along with supervision of compliance, incident reporting and investigation, and emergency planning and response procedures (see more possible procedures in the Radiation Protection Program Development Checklist section).

Gamma device storages and the inside of any shielded enclosure should be designated as controlled areas and identified with the radiation symbol; the area immediately outside the enclosure and storage, neighboring corridors, and *X*-ray generator storage rooms, should be designated as supervised areas. The facilities in

which the radiation sources are stored or used temporarily on the client's site should be designated as provisional controlled areas.

Temporary jobsite radiography work should be performed in a cordoning off area designated as controlled area, which should be under continuous supervision by radiographer's assistants. Its boundary is typically set at a dose rate $\leq 20$ $\mu Sv\ h^{-1}$, and delimited using temporary barriers and notices showing the radiation symbol. When practicable, audible signals—a siren, whistle or bell—can be used to warn of the exposure start, and visual signal—flashing beacons—to indicate that the exposure is in progress.

According to experience, a constraint of 5 mSv a year is recommended for occupational exposure of radiographers and assistants; although, an investigation level of 2 mSv could also be established to trigger an examination of the causes and circumstances of the exposure.

All radiography staff is subject to individual monitoring. It is an undeniable fact that whatever a failure to follow procedures occurs while working with projecting exposure devices, there is a big chance to somebody to receive high doses of radiation; therefore, a dosimetry service should be arranged with an accredited organization, and staff provided with thermoluminescent dosimeters (TLD) or film badges. Additional immediate readout electronic personal dosimeter and alarm monitors are recommended when radiographing at temporary jobsites.

Although gamma sources used in industrial radiography have a very low risk of contamination due to the double encapsulation of the radioactive source and their certification as special form, leak test is to be performed to all sources in use at intervals not to exceed 6 months, before use if were stored for a long period, and before its transfer. Exposure devices using depleted uranium (DU) shielding should also be tested for DU contamination at a 12 months interval [11].

Locations where radiation sources are stored or permanently used should be checked and the source presence recorded every working day; a physical inventory of all radioactive sources should be performed quarterly [11]. The radiography organization should also have implemented utilization logs, that is, a book or record describing the make, model, and serial number of radiation machines and/or exposure devices; storage container or source changer in which any sealed source is located; identity and signature of the radiographer to whom it was assigned, and location and dates of use, including the dates removed and returned to storage. For fixed radiography machines, the dates each radiation machine was energized should be recorded.

Workplace monitoring is to be regularly performed in shielded enclosures—in operator's position, around the walls, doors and openings, and in adjacent occupied areas—and around the radioactive source storage; also, when radioactive sources are renewed, or any change in radiography techniques or beam direction is made. Workplace monitoring is also frequently performed in supervised areas. The dose rate at the entrance to shielded enclosures should be measured after finish the work to confirm that the gamma source has been satisfactorily returned to the exposure device or that *X*-ray emission has stopped.

At temporary jobsites, workplace monitoring is usually performed around the barriers delimiting the controlled area to confirm they are correctly positioned; at the operator position when energizing the *X*-ray generator or ending the exposure, or when winding-in and winding-out the gamma source; around the exposure device after finish the work, to confirm that the source has been fully returned to the shielded position; around the jobsite to confirm that no sources have been left on it; around the on-site source storage to check it is properly shielded; and around vehicles prior gamma sources departure to and from the jobsite [12]. Radiographers, assistants, and radiation safety officers should be provided with sufficient portable measuring instruments, capable of measuring gamma and *X*-ray radiation in a wide range (20 μSv $h^{-1}$–10 mSv $h^{-1}$) [11]. These instruments are to be calibrated at the corresponding energies at intervals not to exceed 6 months or after instrument servicing, except for battery changes.

It is also important to consider an adequate program—initial, refresher, and on-the-job—for training radiographers and radiographer's assistants, managers, and radiation safety officers. Main topics for general training are: basic principles and methods for protection, the use of measuring instruments and units of measurement, actions to be taken in an emergency, applicable regulations, local rules, and operational procedures, etc. Industrial radiographer training include 2 months of on-the-job training, and a radiographer certification program by a certifying entity [10]. NRC's certifying entity is the American Society of Nondestructive Testing (ASNT). Radiographers and radiographer's assistants refresher training is to be scheduled yearly and acquired knowledge demonstrated by written tests and practical examinations.

Certainly, the radiation protection program should also consider its annual review. This annual review is to confirm that the program meets all regulatory requirements, is adequate to ensure the source, workers and public safety, and has been followed according to the expectations. The review should be also to analyze the program faults and possible improvements.

Regarding recordkeeping, a copy of the license and its amendments, as well as radiation machine registrations, authorization conditions, and proofs of receipt and transfer of radiation sources should be kept until the license or registration is terminated and until 3 years after its transfer or disposal. Records of alarm system and entrance control checks at shielded enclosures, inspection and maintenance of radiation machines, radiographic exposure devices, transport and storage containers, source changers, and survey instruments should be kept for 3 years from the date of the inspection or maintenance. The results from leakage tests, DU contamination, workplace monitoring surveys and portable instrument calibrations, should also be recorded and the records kept for 3 years from the date of the calibration or measurement. Records of personal training attendance and certification are maintained for 3 years after the record is made. Problems found in daily checks and quarterly inspections, and the results from quarterly inventory and individual monitoring should also be properly recorded and documented.

Emergency response procedures should, at a minimum, contain the following elements: steps for retrieving a jammed source, including its previous training using

a dummy source; use of remote source handling tools; access restriction to the vicinity of the source; communication to the organization RSO; availability and location of emergency shielded containers, remote tools, and measuring instruments. These plans should clearly give details of any external response, in particular, the arrangements to receive external help, especially for people who can be injured or overexposed.

An industrial radiography facility should be formally decommissioned. This means that there should be a decommissioning program including the transfer of all gamma sources, containers (exposure devices) made of DU, and *X*-ray generators to another authorized organization, or disposal operator, or returned to the manufacturer or distributor, as appropriate. A comprehensive final monitoring should be conducted to confirm the absence of contamination and any radiation source after decommissioning; the corresponding documentation and reporting should be submitted to the NRC or Agreement State. Any *X*-ray generator that cannot be transferred should be made inoperable.[2]

## 17.3 Nuclear Gauges

Nuclear gauges have been used for many years to control liquid and solid levels, material thickness, density, flow, moisture, and other parameters in several industries, including automobile and aircraft manufacturing, oil and gas, paper and plastics, cement and concrete, etc. They operate on the principle of radiation attenuation, i.e., the difference between the emitted radiation and the received radiation, to quantify the desired parameter. Most frequently used sources are $^{85}$Kr, $^{90}$Sr/$^{90}$Y, and $^{147}$Pm as beta emitters, $^{55}$Fe and $^{109}$Cd as *X*-ray emitters, and $^{137}$Cs, $^{241}$Am, and $^{60}$Co as gamma emitters. Neutron sources of $^{241}$Am-Be and $^{252}$Cf are used for measuring soil and asphalt moisture content, and neutron generators, to identify the presence of specific atoms. Source activity varies in a wide range from tenths of GBq to some tens or hundreds of GBq. Examples of nuclear gauge applications can be found in Table 3.3.

The radioactive sources used in fixed nuclear gauge generally fall under IAEA source category 3, while sources of most of portable gauges fall under category 4 [7].

Nuclear gauges can be fixed and portable. Fixed gauges are typically used in mines, mills, and industrial processes; portable gauges are used in road construction, and agricultural and forestry settings. Nuclear gauges usually consist of a suitable radiation source with a shutter to prevent access to any areas of high radiation, and one or more detectors; they are classified into three groups:

[2]An inoperable radiation machine is one that cannot be energized when connected to a power supply without repair or modification.

- Transmission gauges, which measure radiation that passes through the material; the source and the detector, are on opposite sides;
- Backscatter gauges, which measure radiation backscattered by the material; the detector and the source, are mounted on the same side; and
- Reactive gauges, which measure the fluorescent *X*-rays of characteristic energy caused by the ionization of specific atoms; it indicates not only the presence of specific atoms, but also its amount in the material.

The NRC, or an Agreement State, licenses the possession and use of portable and fixed gauges [16, 17] with sources registered in the Sealed Source and Device Registry (SSDR). Gauging devices that incorporate a neutron generator require a Type A specific license of broad scope [18] and the particle accelerator is subject to individual State registration as a radiation producing machine [4]. The organization requesting to install and use nuclear gauges, should perform a safety assessment for a range of scenarios representing normal use and reasonably foreseeable incidents, including contamination from a damaged source scenario of very low probability [19].

The radiation protection program should have an initial statement outlining the safety policy and accountability for safety, including the responsibilities and authorities of the corresponding levels of management reflected in the organizational chart. Such statement may appear like a paragraph indicating the safety priorities and program applicability. It is also necessary to identify all the radiation sources at the organization, their technical specifications and locations. It is also appropriate to refer to each source license (or registration) number and docket, and the main regulations that served as the basis for the radiation protection program.

A radiation safety officer (RSO) and a senior manager with authority regarding radiation safety should be appointed at the organization using fixed and portable nuclear gauges. If the work is carried out outside the main organization site, an adequate number of radiation protection officers should be designed to be available at client jobsites. In medium to large sized organizations, it may be recommended to have a radiation safety committee to regularly review the performance of the radiation protection program. The radiation safety committee may include the RSO, environment and health safety (EHS) officer, maintenance engineer, security officer, senior manager responsible for radiation safety, and representatives of the workforce.

Since members of the public include persons who work, or can be near locations where nuclear gauges are mounted, the most effective way to limit the dose to the public is to prevent these people from entering operational areas during gauging work; permanent and temporary controlled areas should be established. For example, when working with portable devices, any area where the dose rate could be $> 7.5$ $\mu Sv\ h^{-1}$ is designated as a temporary controlled area, delimited with physical barriers—cordon or tape barrier, barricade, etc.—and signaled with a radiation warning symbol. Provisional storage locations, e.g., storage shed, or the

trunk of a car, should be also designated as a temporary controlled area, secured using keyed locks, padlocks, etc., and signaled with the radiation warning symbol.

Fixed gauges are those permanently installed in the production line, e.g., on a pipe, chained and locked to a storage rack or secured to a metal framework, mounted against the inside wall of a vessel, etc.—or used at temporary jobsites, e.g., mounted to vehicles or trailers, or temporarily installed on process equipment; thence, controlled areas are largely restricted by the source housing, its mechanical guarding that also serves as physical barrier, and automatic shutters. Lockout procedures—locking the shutter into the OFF position and tagging the shutter control mechanism—are required during maintenance, repair, or work in, on, or around the process line where these devices are mounted.

Portable gauging device storage rooms (or space, closet, etc.), and stores for gauges awaiting installation, gauges or containers with spent sources for disposal or return to the manufacturer, are usually permanent controlled areas too; depending on the measured dose rates, the area immediately outside storage room and neighboring corridors might be designated as supervised areas.

Permanent controlled and supervised areas should be drawn in a map indicating the fixed nuclear gauge locations, their type and use, adjacent areas and nearest full-time workstations, along with the established points for routine workplace monitoring. Workplace monitoring is to be routinely performed around any source storage facility, including portable storage spaces, and any other source storage in jobsites; around barriers if performing portable gauging operations to establish the extent of the controlled area; at the operator position during source loading/unloading, or when neutron generator is energized/terminated; around the site on completion of a gauging work; and around vehicles with gamma sources prior to departure to and from the site.

Bearing in mind that fixed gauges are frequently located in harsh environments, it is important to regularly check the levels of radiation on and around the gauge for both the beam ON and OFF conditions, and before carrying out maintenance on or close to a gauge, to confirm the shielding is intact. Nuclear gauges are also to be regularly leak tested by smearing the inner accessible surface of the gauging device, source changer or container according to standard methods [20] at intervals not to exceed 6 months [17]. Besides, since it is possible for neutron generators to have some radioactive contamination on outer surfaces, contamination should also be checked in these locations at least once a year, upon receiving or shipping out a generator, and following a repair [21].

Another important part of the radiation protection program is to provide for the sufficient quantity of radiation monitoring instruments, including neutron and contamination rate meters, if appropriate, and their calibration—at least once a year, and after any servicing or repair—by an accredited organization.

Individual monitoring is only required for maintenance or repair personnel who performs nonroutine tasks, e.g., gauge installation, relocation, replacement, alignment, or removal from service; and are likely to receive a dose $\geq 1$ mSv in a year. Gauging operations involving neutron sources or neutron generators require, in their turn, the use of individual neutron dosimeters. In portable gauging work,

individuals likely to receive a dose $\geq 1$ mSv—those whose work involves detaching the source or source rod from the device—may be also required to wear personnel dosimeters. An arrangement with an accredited dosimetry service using film badge or thermoluminescent dosimeters (TLD)—and/or neutron dosimeters optimized for the effective neutron energies—should be then considered and implemented. Occupational exposure of workers not subject to individual monitoring is to be evaluated based on workplace monitoring. Personal alarm monitors are also recommended on a supplementary basis.

Procedures and local rules should cover normal gauge operation, lockout/tagout to prevent shutter or beam opening during maintenance or repair works, selection and delimitation of controlled and supervised areas, routine and nonroutine maintenance operations, shutter mechanism testing, new gauges procurement, source exchanges, unauthorized access or removal of fixed gauges, gauge storage withdrawals and transportation to and from temporary jobsites, measuring instruments using and testing, disposal/shipping of spent radioactive sources, source leak testing, etc., along with procedures for compliance supervision, audit, incident reporting, and emergency planning and response (see more possible procedures in the Radiation Protection Program Development Checklist section).

An adequate training and experience program—initial, refresher, and on-the-job—should also be planned. There are courses available for RSOs and nuclear gauge authorized users [22, 23] accepted by regulations [16, 17, 24]. Individuals responsible for radiation safety should typically complete a course for users (initial gauge safety training and a HAZMAT course), an 8-hour radiation safety course, and 8 h hands-on experience with fixed gauges. Written or oral examinations and observations are required. Refresher training is to be scheduled at least every 3 years.

Practical exercises associated with procedures for retrieving a stuck gauging source—without a real source—and additional training associated with, e.g., shutter, source drive mechanism, or shielding repair and maintenance, and leak testing are also required. Personnel who work in the vicinity of the nuclear gauge should be at least instructed on the gauge nature and location, the meaning of the radiation symbol, not touching the gauge and keeping away from it as much as their work permits.

Certainly, the radiation protection program should also consider its annual review. This annual review is to confirm that the program meets all regulatory requirements, is adequate to ensure the source, workers and public safety, and has been followed according to the expectations. The review should be also to analyze the program faults and possible improvements.

Regarding recordkeeping, a copy of the license and its amendments and conditions, as well as any source certifications, should be kept until the license is terminated. Documents of receipt and transfer of nuclear gauges and neutron generators are to be maintained for as long as the material is possessed and until 3 years after its transfer or disposal. Records of the program content and modification, as well as audits and reviews of its implementation, are to be kept, as a minimum, 3 years from the date the modification, audit, or review took place.

Records of personal training, gauge, and neutron generator maintenance and inspection, survey instrument maintenance and repair, results from leakage tests, workplace monitoring results, including contamination, if appropriate, and survey meter calibrations, etc., should also be kept for 3 years after the record is made.

Locations where portable and fixed gauges are stored, and fixed gauges permanently installed and used at temporary jobsites, should be checked regularly to ensure that the sources are present; sources in permanently installed fixed gauges are usually accounted for at least once per month; while sources in portable gauges are accounted for daily using a log book if out of the store and, weekly, when in storage. The log book should include the date of use, the name of the authorized user, and the temporary jobsite where the gauge will be used. A physical inventory of all sources should be conducted every 6 months. Any loss, theft, or unauthorized removal should be immediately reported.

Emergency response procedures should analyze what the consequences are and what to do in scenarios such as a stuck source or stuck shutter; dropped or leaking source; source holder shielding damage caused by a fire or natural disaster; and missed or stolen gauging devices. Emergency procedures include actions, such as stop using the gauge, evacuate the workers in the vicinity, delimit the affected area and restrict its access, notify relevant authorities, including the organization RSO and the person responsible for safety. The availability and location of emergency shielded containers, remote tools, and measuring instruments should be part of the plan; also, the arrangements to receive external help, especially for people who can be injured or overexposed.

Nuclear gauges should be formally decommissioned. This means that, when nuclear gauges and neutron generators are no longer needed, the organization should implement its transfer to other authorized organization, or disposal operator, or return them to the original manufacturer or distributor. Old-design gauging sources which original manufacturer or distributor is no longer in business, are to be treated as orphan sources and transferred to a disposal operator. If a neutron generator cannot be returned to the original manufacturer, it should be made inoperable before transferring it to a disposal operator. The decommissioning program should also consider a comprehensive final monitoring to confirm the absence of contamination, and any radiation source at the site, and the submission of the corresponding documentation to the NRC or Agreement State.

## 17.4 Well Logging

Well logging organizations use instruments lowered into a well—a hole drilled in the ground—to obtain information about the geological properties, e.g., density, porosity, and hydrocarbon content of the rock formation around the well; measure drilling parameters, such as fluid temperature, pressure, and flow rate; and detect casing corrosion, wear and other equipment damage. The instruments, which can bear one or more radiation detectors, radioactive sources and neutron generators,

are lowered into the well on a cable known as a wireline. The wireline carries the signals from the logging instruments to the surface, where they are recorded. As the wireline tool is slowly raised, the log plots the parameter being measure against the depth [25].

Nuclear well logging tools include various measurements techniques. The first is a gamma measurement technique that uses no source, which simply measures and identifies naturally occurring gamma rays to help distinguish the shale content of sedimentary rocks and aid lithological identification. A second technique provides a neutron-neutron or compensated neutron log from a 4−5 MeV neutron radioactive source of up to several GBq of $^{241}$Am-Be or $^{238}$Pu-Be to indicate how porous the rock is, and whether it is likely to contain hydrocarbons or water. $^{252}$Cf sources seem to be comparable and can be potentially used as well. A third technique uses a gamma-gamma or density tool containing a $^{137}$Cs, with an activity usually of up to 75 GBq. The amount of gamma backscatter from the formation provides the density log that, together with the porosity log, is a valuable indicator of the presence of gas. A fourth technique, called neutron-gamma logging, houses a linear accelerator and a $^{3}$H target of several hundred of GBq to generate 14−15 MeV neutrons. Elements are identified measuring the gamma radiation emitted by the activated atoms or the thermal neutron decay characteristics.

Technologies that allow well logging during drilling—logging-while-drilling (LWD) and measurement-while-drilling (MWD)—by incorporating the logging tools in the drill collar or coiled tubing, additionally require small radioactive sources referred to as energy compensation sources, or ECS, to stabilize and calibrate the well logging tools. These are low-activity special form, singly or doubly encapsulated sources with an activity $\leq 3.7$ MBq. While running the casing, it is also normal practice to insert small radioactive sources to act as depth correlation markers; they provide clear indications on the log of when the logging tool reaches the defined depths. Tag sources each contain about 50 kBq of $^{60}$Co in the form of malleable metal strips or pellets.

Well logging can also involve the use of unsealed radioactive sources for tracer, field flood or labeled frac sand studies. In these studies, one or more radionuclides in liquid, solid, or gas form, or chemically bonded to glass or resin beads, are injected into a single or multiple wells to monitor the movement of fluids or gases—tracer studies—or to determine the direction and rate of flow through the formation—field flood studies—or to assess the amount of radionuclide remaining in the underground reservoir formation—labeled frac sand studies. The work with unsealed radioactive sources requires appropriate containment and washing facilities, and the use of personal protective equipment to prevent contamination.

Radionuclides used in these studies are $^{3}$H, $^{14}$C, $^{22}$Na, $^{35}$S, $^{45}$Ca, $^{46}$Sc, $^{59}$Fe, $^{60}$Co, $^{63}$Ni, $^{65}$Zn, $^{82}$Br, $^{85}$Sr, $^{85}$Kr, $^{90}$Sr, $^{110m}$Ag, $^{124}$Sb, $^{125}$I, $^{131}$I, $^{140}$La, $^{192}$Ir, $^{198}$Au [26]. They are selected so as to be consistent with the materials to be studied and their decay characteristics, e.g., radiation emission, half-life, low initial activity, etc.

Sealed sources and accelerators used in well logging, generally fall into IAEA source categories 2, 3, and 4 [7]. NRC or an Agreement State should approve a specific license for the use of byproduct materials, source materials, and special

nuclear materials for well logging, including well logging and tracer applications involving both single or multiple well bores, conventional well logging and tracer operations, although special authorizations may be additionally required. The license authorizes the use of energy compensation sources (ECS) in the tools with an activity not exceeding 3.7 MBq, and particle accelerators using tritium target sources as neutron generators [27, 28]. The particle accelerator is also subject to individual State registration as a radiation producing machine [4].

To perform well logging services, well logging organizations are required to sign a written agreement with the well owner or operator. This requirement is not applicable if the licensee and well owner or operator are part of the same corporate structure or otherwise similarly affiliated. This written agreement is to define the responsible for the efforts to recover a source if it becomes lodged; for continuously monitor the circulating fluids in the well bore during recovering; and, if the source is declared irretrievable, for obtaining approval and implement the abandonment procedures—immobilize the source and seal the well to prevent inadvertent intrusion by subsequent drilling operations [28].

Well logging organizations should also perform and document a safety assessment involving each and all radiation sources for which they are seeking authorization. The assessment should include potential exposures to well logging engineers, other workers, and the public, for a range of scenarios representing normal use and reasonably foreseeable accidents and incidents, with both shielded and unshielded radioactive sources and electrical neutron generators including contamination from a damaged source scenario of very low probability [29]. A possible spillage and release of radioactive material from pressurized systems, and unintended or unauthorized disposal of waste, is also to be assessed if unsealed radioactive sources are to be used.

The radiation protection program should have an initial statement outlining the safety policy and accountability for safety, including the responsibilities and authorities of the corresponding management levels reflected in the organizational chart. Such statement may appear like a paragraph indicating the safety priorities and program applicability. It is recommended to identify here all well logging sealed sources, neutron generators, tritium target sources, radionuclide tracer materials, source materials, and depleted uranium that are used, with their technical specifications and locations. It is also appropriate to refer to each source license (registration) number and docket, and the main regulations that served as the basis for the radiation protection program.

A radiation safety officer (RSO) and a senior manager having authority regarding radiation safety should be appointed. A sufficient number of radiation protection officers should be designed to be available at well owner or operator sites. It is also recommended to have a radiation safety committee to regularly review the performance of the radiation protection program. The radiation safety committee may include the organization RSO, environment and health safety (EHS) officer, drilling manager, well logging engineer, security officer, senior logging supervisor, and representatives of the workforce.

For cased and uncased boreholes, sealed sources and neutron generators are typically located in source holders in the logging tools. Each logging tool should be permanently and clearly marked with the radiation symbol; a warning notice with the word RADIOACTIVE; and the source radionuclide and activity for which the tool is suitable. If a sealed source becomes lodged in a well, as it may occur, all reasonable efforts should be made for its recovering; but if the source is classified as irretrievable, it must be immobilized and sealed in place with a cement plug, and the site identified with a plaque bearing the word CAUTION, the radiation symbol and all necessary data to identify the well and the source.

If a source becomes stuck in a source holder, it is possible to remove it using destructive techniques, e.g., drilling, cutting or chiseling; though, these operations can generate radioactive contaminated materials, and should be specifically approved by the NRC or Agreement State.

Sealed sources are removed from the tool for maintenance, repair, transportation, and calibration. When the source is unloaded from the well logging tool, it is then loaded into a shielded container for storage. The shielded container is designed to reduce the radiation levels from the source and prevent any unauthorized access. During unloading and loading, the source is exposed for a short period of time, and the dose rate can exceed 7.5 $\mu Sv\ h^{-1}$ for up to 30 m in the forward direction and about 4 m behind the engineer; the dose rate from a neutron source can exceed 7.5 $\mu Sv\ h^{-1}$ at distances of up to about 4 m [30]. Any work that includes the removal of radiation sources requires appropriate remote handling tools—the logging tool may also have a source handling tool—local shielding, as practicable, and the use of barriers to designate the extent of the controlled area. Demarcate a controlled area can be a challenge in tight spaces, for example, on offshore production platforms; in such cases an overpack in the form of a large thick-walled box is used as a temporary store, and constraints are established for these operations.

Source changers are used for the safe exchange of old and new sources from logging tools. A source changer is a shielded container equipped with an additional empty channel, coupled to a source projector for the exchange [31]. Source changers incorporate a system for ensuring that the source is not accidentally withdrawn when connecting or disconnecting. They also include a lock or have an outer locked container designed to prevent unauthorized or accidental removal of the source from its shielded position. Normally, the source changer is the shipment container and should be returned to the supplier.

Neutron generators emit gamma radiation during neutron generation and also for some time after the generator is turned off. They can have in addition some radioactive contamination on the outer surfaces; hence, a holding time for the decrease of activated activity in tube parts is necessary, along with the use of appropriate personal protective equipment, and contamination monitoring instruments.

The most common use of radioactive tracers is at the well head; the radiotracer logging tool that carries the radioactive material has a reservoir that can be selectively released into a flow stream from a “breaker sub”—a test tube which explodes at the appropriate depth with a small electric blasting cap or squib—or from an

injector tool, a motorized device capable of ejecting controlled radiotracer quantities when activated from the surface. Adequate containment for actual and potential contamination, personal protective equipment, and contamination monitoring instruments should be available, as well as washing facilities to clean the areas and decontaminate tracer injection tools, other equipment, trucks, and laundry, if appropriate.

Radiotracers are usually acquired in pre-calibrated amounts—injections—or "ready to use" quantities; yet, some engineered controls, such as a hood or an extract ventilation system, may still be needed at the well head for "repackaging" gaseous, volatile, or finely divided tracer material. Otherwise, radiotracers have to be prepared in a laboratory by labeling adequate amounts of non-radioactive material—water, oil, lubricator, or gas—with the radionuclide. The laboratory should have appropriate facilities, such as controlled areas equipped with fume hoods, and extract ventilation systems for handling unsealed radioactive sources, and prevent the dispersal, ingestion or inhalation of radioactive material.

Radiotracer work also generates radioactive waste that should be collected and disposed of, including laboratory aprons, gloves and overshoes, absorbent materials, glassware and similar low level radioactive waste, surplus radiotracers and waste water from decontamination facilities. In this case, the radiation protection program should consider, whenever possible, the use of relatively short lived radionuclides, minimum quantities of radioactive material, strict controls to minimize contamination and the return of spent sources, unused source materials and byproduct materials to the manufacturer to reduce the amount of radioactive waste.

Since members of the public include workers and other employees at the vicinity of the jobsite, a way to limit public exposure is to properly designate controlled areas the locations where well logging operations are conducted and sealed sources and tracer materials are stored; controlled areas are particularly important during well logging source loading and/or unloading in the tool and calibration, and when carrying out radiotracer procedures. Controlled areas are to be delimited with physical barriers—cordon or tape barrier, barricade, etc.—signaled with the radiation warning symbol and visual signals—e.g., flashing beacons—and be constantly supervised to ensure that no authorized persons enter the area. Loading/Unloading operations should be conducted quickly; and no other work can be permitted in the area until the work with the radiation sources has been finished. It is critical to bear in mind that well logging work can be in an oil field or an offshore platform, where the space might be limited. In such cases shielding is always favored to reduce the extent of the controlled area.

Permanent and temporary controlled areas should be drawn in a map, including, as appropriate, operational areas, radiotracer laboratories, lockable stores for sealed radioactive sources and tracer materials, well logging tools, spent sources and radioactive waste storages and any below ground bunker storage or container (downhole storage), well logging trucks storing radioactive sources, and laundry and decontamination facilities, if any, specifying how the contaminated waste water from laundry machines or sinks is disposed. Besides specifying the shielding materials, if any, and the relative distance between restricted areas and nonclassified

areas, the drawings should indicate the established points for routine workplace monitoring. The surrounding areas should be described as well and might be designated as supervised areas, as necessary.

The monitoring program should describe the monitoring frequency for dose rate, surface contamination, and airborne contamination. Workplace monitoring is to be routinely performed around storage facilities; around barriers during well logging operations; at the operator position when the sources are used or the neutron generator is energized down the hole; at the operator position during a source loading/unloading; around the site before start and on completion of a well logging work; and around vehicles transporting or storing radioactive sources, including the driver seat. A dose rate survey should also be conducted on completion of the well logging work to ensure that all sources have been placed into the shielding containers and no sources have been left in the tool or have become detached.

Contamination monitoring should be regularly performed on the surfaces of the tool string, well logging operational area, neutron generator outer surfaces, and transport containers. Neutron generators contamination should be checked at least once a year, upon receiving or shipping out a generator, and following a repair [21]. The program should also provide for a routine skin and clothing contamination monitoring if working with radiotracers. A decontamination program should also be in place to deal with decontamination of work areas, equipment, vehicles, and unrestricted areas.

The radiation protection program should also provide for the adequate quantity of radiation monitoring instruments, including neutron and contamination rate meters, and for instrument calibration—at least every six months, and after servicing or repair—by an accredited organization.

The inspection and maintenance of source holders, logging tools, source handling tools, storage containers, transport containers, and injection tools, at intervals not to exceed six months, should also be considered in the radiation protection program to assure their proper labeling and physical condition. Sources are to be moved only in shielded containers, containers should be locked and the keys removed and kept only by authorized persons. If a vehicle or trolley is used to move a container, it should be securely fastened inside the separate compartment of the vehicle.

Source leak testing services are also required for sealed sources containing greater than 3.7 MBq of beta/gamma or 0.37 MBq of alpha radioactive material at intervals not to exceed 6 months, and at intervals not to exceed 3 years for energy compensation sources (ECS) [26].

Well logging personnel should wear film badges or thermoluminescent dosimeters (TLD) when handling, transporting, calibrating, and assembling well logging radioactive sources, and/or working in controlled areas. Optically Stimulated Luminescence (OSL) may also be authorized [27]. Hence, arrangements should be made to receive individual monitoring services, including neutron dosimetry, from an organization accredited by the Laboratory Accreditation Program (NVLAP). Additional personal alarm monitors can be used only to supplement the personal dosimetry service. A bioassay program to control internal

exposures may be also necessary if handling $^{3}H$ in excess of 3.7 GBq, or gaseous $^{3}H$ in excess of 3,700 GBq, or using radioiodine radiotracers with more than 1.85 MBq at any one time, or a total of 1.85 MBq within any 5 days period [26].

Procedures and local rules should cover the abandonment of an irretrievable well logging source; routine well logging operations, including source receipt, exchange, preparation to ship to well site, and the checks before logging; safe loading/removal of the source in/from the tool; the use of remote handling tools; source storage and accounting; periodic source physical inventory, source movement and record-keeping; controlled and supervised areas selection; routine and nonroutine logging tools maintenance and calibration; object and person decontamination; measuring instruments using and testing; handling and specific uses of tracer materials; handling and use of accelerator targets or tubes containing radioactive materials; bioassay samples collection; disposal/shipping of spent radioactive sources; segregation, collection, and disposal of radioactive waste, including waste water from laundry machines or sinks; etc., along with procedures for compliance supervision, incident reporting, and emergency planning and response. Any relevant investigation level or authorized level should be considered, as well as the procedures to be followed in the event that any such level is exceeded (see more possible procedures in the Radiation Protection Program Development Checklist section).

The radiation protection program should also contemplate the frequency and content of the — initial, refresher, and on-the-job—personnel training. The RSO should be qualified by training and experience, i.e., should have an academic degree in science, board certification, and years of experience, or formal training, in the conduct of a radiation safety program. Logging supervisors, logging assistants, and individuals authorized to conduct field flood studies should have received initial training in subjects, such as fundamentals of radiation safety; radiation detection instruments; remote handling tools; regulations and license requirements; and case histories of accidents in well logging. 520 h on-the-job training is required to demonstrate competence in the use of well logging tools, remote handling tools, and radiation survey instruments for gas or oil well logging operations, and 160 h, for mineral logging [28]. Field flood studies additionally require at least 8 h of classroom training for tracer studies. The understanding and skills are to be demonstrated by successfully completing a written test and a field evaluation. Refresher training courses should be scheduled annually.

A semiannual physical inventory of all sources—sealed sources, tritium targets, depleted uranium, and radiotracers—and an annual inventory reconciliation, if applicable, are required to confirm that the sources are in their assigned locations or were used or taken out from the inventory according to established procedures. To maintain the inventory updated, authorized, and trained workers are required to sign out the sources removed from store or moved to another location in a log book, including the movement date, name of the authorized user, and the jobsite where the source was moved or transferred. Well logging tools that incorporate neutron generators should be included in accountancy procedures. A detailed waste inventory, including waste type, radionuclide and activity, as well as all sources

removed from regulatory control or transferred to other facilities, should also be maintained.

Certainly, the radiation protection program should also consider its annual review. This annual review is to confirm that the program meets all regulatory requirements, is adequate to ensure the source, workers and public safety, and has been followed according to the expectations. The review should be also to analyze the program faults and possible improvements.

Regarding recordkeeping, copies of the licenses and its amendments and conditions, as well as all additional authorizations should be kept until licenses are terminated. Documents of receipt and transfer of all sources and neutron generators are to be maintained for as long as the material is possessed and until 3 years after its transfer or disposal. Records of the program content and implementation, its audits and reviews, are to be kept for 3 years from the audit or review date. Other records that should be kept for 3 years include personnel training and certification, source physical inventories, leak test results, waste collection and disposal, releases of licensed material into the sanitary sewerage, field flood injections, identification of wells, uses of radioactive material and radiation sources, equipment defect inspection, labeling inspection, survey instruments maintenance and repair, instrument calibration, workplace monitoring results including contamination, etc.

When logging tools are placed in a well, it is possible that the wireline support breaks or the tool becomes "snagged" within an open hole—uncased—and, regardless of all efforts for "fishing" the disconnected logging equipment, the radiation sources may not be recovered. As a result, the integrity of the radioactive material encapsulation may be damaged, and a widespread radioactive contamination of the wellbore, drilling rig, fishing tools mud tanks, mud pumps, and other equipment that comes into contact with the drilling fluids may occur. Hence, emergency preparedness and response procedures should analyze scenarios, like extended exposures of personnel due to a difficult removal of a source from the logging tool or improper handling of well logging sources; contamination and overexposures as a result of physical damage to the sources, containers and other equipment by mechanical, thermal or chemical means; missed, dropped, lost or leaking sources; and a fire and explosion owing to the highly combustible products of the oil and gas industry; natural disasters and malevolent acts.

Emergency procedures should include communication and coordination arrangements with the well owner/operator and external emergency response organizations—NRC or agreement state authorities, firefighters, health practitioners, etc.—and address actions to evacuate the workers in the vicinity, demarcate the affected area and restrict the access, protect emergency workers, decontaminate the area around wellhead and any equipment used in the recovery operations, and classify, account and store the resulted radioactive waste. The availability and quick location of emergency spare shielded containers, remote handling tools, other tools as appropriate, and an adequate quantity of measuring instruments and dosimeters to deal with the potential radiation exposure of firefighters and other personnel should also be part of the plan.

Well logging service facilities should be formally decommissioned. This means that the program should anticipate the transfer of all gamma and neutron sources, and neutron generators, to other authorized organization, or disposal operator, or be returned to the original manufacturer or distributor, when the organization permanently ceases its operations. Old-design well logging sources, which original manufacturer or distributor is no longer in business, are to be treated as orphan sources and transferred to a disposal operator. If a neutron generator cannot be returned to the original manufacturer, it should be made inoperable before transferring it to a disposal operator. The decommissioning program should also consider a comprehensive final monitoring to confirm the absence of contamination and any radiation source at the site. The operating organization should keep records of all authorizations for the receipt, storage, transfer or disposal of radioactive sources, and submit this documentation when reporting the decommissioning.

## 17.5 Custom and Border Inspection

For a long time, human screening for security or antismuggling purposes has been deemed not justified; however, the development and availability of systems capable of producing images with extremely low doses of radiation, and the need for an increased security in response to the rising terrorist threats, have had an effect in the justification of these practices. Out of necessity, the inspection imaging device[3] developments of the last decade are impressive. *X*-rays are now massively used in baggage inspections systems installed at the entrance of public buildings and airports; post-room scanners for screening mail, small parcels and small bags for bombs, chemicals, weapons and drugs; border checkpoints vehicle screening systems for scanning passenger cars, vans, buses, and mini buses; portable *X*-ray units for on-site inspection of suspicious objects or cargo; backscatter vans to screen drive-by vehicles for explosives, weapons and drugs; and low dose rate backscatter units for the detection of organic threats, contraband and explosives.

Mobile scanners used for cargo inspection at ports, airports, truck terminals, and for scanning cargo/tank wagons, or containers loaded onto trains, are increasingly improved with the use of interlaced dual energy accelerators. In some systems both the subject and the system are stationary, in some the subject moves through the system, and in some the system moves past the subject. Drive-through systems typically operate with 3−6 MeV energy, and train scanning systems with 6−9 MeV energy for high passing speeds.

Fixed and relocatable cargo scanners also use more penetrating $^{137}$Cs and $^{60}$Co gamma sources—gamma gauges—as inspection imaging devices, that need to be kept in a shielded enclosure at all times. The exclusion zone is typically surrounded

[3]Inspection imaging device is used to generalize all systems specifically designed for imaging persons or cargo conveyances to detect concealed objects.

by radiation shielding with interlocked doors—or a fence with interlocked gates—to maintain the dose rate at 0.05 μSv/h outside it. The Vehicle and Cargo Inspection Systems (VACIS) developed for the nonintrusive inspection of goods in transportation systems, use gamma ray technology to detect weapons, contraband and other items of interest [32]. The gamma beams are shuttered open to allow imaging of the container, or cargo portion of a truck, or moving railcars. The software converts the count data into a visual recreation of the material in the container and allows a further enhancement of the image and color variations. The speed of the vehicle during the scan is monitored by a radar gun and used to correct the image distortion due to the variable speed.

Electron capture devices using a $^{63}$Ni radioactive source have also been developed to detect trace quantities of explosives and narcotics; these detectors are based on swab or air sampling of particulate and vapor on or about the person being checked and their belongings.

Regarding nonmedical human imaging, ionizing radiation is used in inspection imaging devices using transmission and backscatter technologies. Backscatter technology uses *X*-rays to detect objects hidden under clothing that can be used for criminal acts in passengers boarding aircrafts, persons crossing a national border, and visitors to prisons. Transmission technology systems uses *X*-ray and gamma sources to detect objects that have been ingested, hidden in body cavities, or implanted under the skin. Generally, the radiation dose to the scanned individual from a backscatter system is much lower than the dose from a transmission system, which is not supposed to be used as a routine screening tool.

Typically, the effective dose from a backscatter *X*-ray is of 0.1 μSv per image of the front of the body, and the exposure is predominately to the skin. Transmission system effective dose per scan ranges from 2 to 5 μSv, depending on the equipment [33]. An individual annual effective dose constraint of 0.25 mSv from a single source—meaning of it all security screenings—has also been recommended based on the general public limit of 1 mSv [34].

According to NCRP recommendations, depending on the potential radiation risk associated with the security screening, human inspection imaging devices are classified in general-use systems and limited-use systems [35, 36]. General-use systems stick to an effective dose of 0.1 μSv or less per scan, and can be used mostly without regard to the number of scans per individual in a year; they would allow the unlikely figure of 2,500 scans per individual without exceeding the constraint of 0.25 mSv a year. Limited-use systems require effective doses per scan greater than 0.1 μSv, but less than or equal to 10 μSv. Limited-use systems require additional administrative controls and should be used with discretion in terms of the number of scans per individual in a year to not exceed the allowed annual constraint of 0.25 mSv a year.

Radioactive sources and generators used for security scanning generally fall into IAEA source categories 2 and 3 [7] and should be kept under proper control. Licensees should ensure that they obtain the radiation sources from authorized suppliers only; also, that they are returned to the original supplier or transferred to an authorized organization at the end of their lifetime.

FDA regulates manufacturers of electronic products that emit radiation, including nonmedical security products using *X*-ray tubes, linear accelerators, or any other electronic source of radiation, and states regulate the use of these products. As any other *X*-ray producing machine, inspection imaging devices must be registered with the FDA Center for Devices and Radiological Health (CDRH), and its operation be authorized by the state agency responsible for radiation protection. Inspection imaging devices with gamma sources should be authorized by a specific NRC's license for the use of licensed material.

The radiological assessment for facilities using inspection imaging devices should consider the exposure of workers who operate the inspection imaging devices, employees who happen to work nearby, screened individuals when imaging persons, and members of the public. The use of such devices may also lead to the inadvertent exposure of people inside a cargo—people smuggling—or inside vehicles being screened or driving vehicles through screening devices. Possible scenarios for potential exposure include flaws in the design of inspection device, failures of inspection imaging devices while in operation, failures, and errors in software that control or influence the delivery of the radiation, and human errors. The inadvertent entry to the controlled area when cargo is undergoing a screening procedure is another way for potential public exposure.

The radiation protection program for custom and border inspection facilities should have an initial statement outlining the organization safety policy and accountability for safety, including the responsibilities and authorities of departments, groups or individuals using the inspection imaging devices. Such statement may appear like a paragraph indicating the safety priorities and program applicability. It is recommended to identify here the involved radiation sources and their type, and to list their main technical specifications and locations. It is also appropriate to refer to each source registration number and docket, license, and the main regulations that served as the basis for the radiation protection program.

A radiation safety officer (RSO) and a manager having authority regarding radiation safety should be appointed at the organization using inspection image devices. These responsibilities should be assigned to cover the entire lifetime of the device, from ordering and receipt, use and storage, to their eventual disposal, sale or other end of life action.

The areas where the inspection imaging devices are used are to be designated as controlled areas; the size and extent of the area are determined considering the occupancy of adjacent areas, doses per scan and workload, system orientation, and people or vehicle traffic flows. For most passenger inspection checkpoints, this could mean clearly demarking the inspection zone—i.e., the area around the personnel security screening system where bystanders are prohibited during the operation of the device—through the use of tape or rope barriers, paint markings on the floor or walls, and signage, including all accesses—ingress, egress, portal, traffic path, etc.—to the area. For large inspection imaging devices including accelerators, high energy *X*-ray generators and some gamma sources, structural shielding may be required—e.g., concrete walls of sufficient thickness—to demarcate the controlled areas. For mobile and relocatable *X*-ray and gamma cargo scanners, the controlled

area includes the exclusion zone, i.e., an area typically surrounded by radiation shielding with interlocked doors or a fence with interlocked gates [37].

By design, inspection imaging devices should incorporate enough safety features to ensure that public exposure requirements are met in areas immediately adjacent to the device: this include precise radiation beam collimation; appropriate shielding; clear warning indications that the beam is open—switched on—and a scan is in progress; captured key switch and interlocked systems to prevent inadvertent exposures; and emergency stop buttons. A 2.5 μSv/h dose rate limit applies to the leakage dose rate at any point 30 cm from any external surface of the device, excluding the beam exit surface, or to the beam exit surface while the shutter is closed or the beam is aligned with a beam stop [36]. Engineering controls are also to be provided to ensure that individuals or vehicles do not reenter the scanning area from the exit.

Individuals operating the inspection imaging device and performing scans, radiation protection officers, and service engineers are occupationally exposed personnel. Personnel controlling lines of entry to the inspection zone are not directly related to the radiation sources and are regarded as members of the public. Since exposures from most of inspection imaging devices are usually predictable and sufficient low—usually the annual dose do not exceed 1 mSv—operators' individual monitoring may not always be required and workplace monitoring would normally be enough. An area dosimeter can be used to confirm the evaluation of occupational exposures. However, this may not be the case for inspection imaging devices using linear accelerators or high activity gamma sources, where the use of individual dosimeters may be necessary. Individual monitoring may be appropriate too for some handheld backscatter units and portable *X*-ray units.

All inspection imaging devices locations are to be drawn in a map of the facility indicating the controlled areas boundaries and the established points for workplace monitoring. Workplace monitoring is to be performed in the immediate vicinity of the *X*-ray unit, accelerator or gamma source, when installation has been completed, and before the device is first used for inspection; when a new software for the inspection imaging device is installed or there is a significant modification to the hardware or software; and after any servicing that may have an impact on the radiation shielding, shutter mechanism or *X*-ray production components has been performed. Annual routine radiation surveys should also be performed. Workplace monitoring results are used to verify public and occupational exposures, and radiation leakage from the device. Besides, a program for the selection, calibration, maintenance, and testing of radiation measuring instruments should be considered.

An inventory of all inspection imaging devices, including procedures for the movement or transfer of security screening systems within the facility, between facilities, to another agency or location, and for disposition of the system is required. The inventory should be reviewed at least once a year.

Local rules and procedures should cover all aspects of inspection imaging device operation and safety, including wearing, handling, and storing of personal dosimeters, if applicable; ambient dose rate investigation levels to trigger an

investigation if they are surpassed; radiation leak checking and workplace monitoring surveys; device and measuring instrument maintenance, calibration and testing; criteria for site selection considering the continuous presence of bystanders and public; recordkeeping; and program periodic reviews, along with compliance supervision, and incident reporting and emergency planning (see more possible procedures in the Radiation Protection Program Development Checklist section).

Inspection imaging device operating and servicing personnel should have appropriate training and experience; though, the program should consider the initial, refresher, and on-the-job training requirements, including the information being provided to the individual being scanned, other safety hazards, and control procedures of the inspection zones. Training should be commensurate with the type of sources used for inspection imaging. A radiation awareness program should also be appropriate for ancillary personnel who work nearby such as security guards, administrative staff, and housekeeping staff. Providing these individuals with basic radiation safety awareness training may prevent misunderstandings and allay some of the fears they may have about working in the area [33].

Records of upgrades, modifications, maintenance, and repair should be maintained for the life of all inspection imaging devices. Other records to be kept include acceptance testing (of devices, instruments, and software); personnel training; individual monitoring and workplace monitoring; incidents and corrective actions; instrument calibrations; source inventories; use, movement and transfer logs, etc. Records of the reference effective dose per screening and the number of individuals undergoing screening procedures each year are required for human imaging, especially for employees or frequent visitors who could receive radiation doses approaching the dose constraint of 250 μSv in a year [36].

Copies of all licenses and registrations, its amendments and conditions, as well as additional authorizations, should be kept until the license or registration is terminated. Documents of receipt and transfer of sources, and neutron generators, are to be maintained for as long as the material is possessed and until 3 years after its transfer or disposal.

Certainly, any radiation protection program should also consider its annual review. This annual review is to confirm that the program meets all regulatory requirements, is adequate to ensure the source safety, and has been followed according to the expectations. The review should be also to analyze the program faults and possible improvements.

Emergency preparedness should also be considered as part of the radiation protection program; emergency procedures should be based on the analysis of reasonably foreseeable incidents—e.g., source loss or theft, or a transport accident when gamma sources are involved; an inadvertent entry to the inspection zone when performing an imaging procedure, etc.—and include measures like the relevant worker training to be able to recognize conditions indicating an emergency situation; the necessary emergency equipment availability and location; and RSO and management notification. Emergency preparedness is particularly important for mobile inspection devices containing radioactive sources that have to be transported.

The radiation protection program should also pay attention to the formal decommissioning of the inspection imaging devices. This means the appropriate disposal of gamma sources and *X*-ray machines by returning them to the original manufacturer or distributor, if it was previously agreed, or their transfer to a disposal operator. Sources or machines can also be transferred to another authorized user. The corresponding documentation and reporting should be part of the decommissioning plan, as well as a comprehensive final monitoring to confirm the absence of radiation sources at the facility. Any *X*-ray generator that cannot be transferred should be made inoperable.

## 17.6 Unsealed Sources

Unsealed radioactive sources are widely used as radiopharmaceuticals for diagnosis and therapy, and, also, as labeled compounds for biochemical analysis. They are extensively used in biomedical research, in neuroscience applications, drug development studies, and to understand different chemical and biochemical interactions in the human body. Unsealed sources are also used to help understand chemical and biological processes in industry, agriculture, and life sciences, e.g., in pipe flow experiments, inter-well studies, plant studies, metabolism research, uptake studies of various components, organism interaction with the environment, etc.

Besides larger nuclear research laboratories, many other research centers might have laboratories for inorganic chemistry, organic chemistry, analytical chemistry, chemical technology, and materials using unsealed radioactive sources with different purposes. Some radionuclides used as radiotracers are $^{3}H$, $^{14}C$, $^{22}Na$, $^{32}P$, $^{51}Cr$, $^{57}Co$, $^{60}Co$, $^{125}I$, and $^{131}I$.

As stated by regulations, any laboratory possessing or using radioactive material in quantities exceeding established exemption limits, should be authorized by a NRC's or Agreement State's specific license for the use of byproduct materials, and should be subject to periodic inspections to confirm that the required conditions are maintained [38–40]. The license is usually bestowed upon completion, among other requirements, of a safety evaluation demonstrating the adequacy of facilities and equipment for the planned use, and the personnel training and experience.

The radiation protection program should have an initial statement outlining the safety policy and accountability for safety, including the responsibilities and authorities of particular departments, groups or individuals, who made use of the unsealed radioactive sources for medical, research or development purposes. Such statement may appear like a paragraph indicating the safety priorities and program applicability. It is recommended to identify here the locations where the sources are used at the facility and their technical specifications. It is also appropriate to refer to each license (registration) number and docket, and the main regulations that served as the basis for the radiation protection program.

A radiation safety officer (RSO) should be appointed at the research facility, hospital or medical center, and the organizational structure for managing the safety use of unsealed radioactive sources should be clearly documented as part of the radiation protection program. A radiation safety committee composed of such persons as the organization RSO, a representative of the environment and health safety (EHS) office, if appropriate, a senior management representative, and persons trained and experienced in the safe use of radioactive materials is a requirement to obtain a Type A specific license of broad scope [39].

The siting and layout of the facility should consider the amount of work with unsealed sources within the nuclear medicine or research facility and, in cases where the laboratory or nuclear medicine unit is part of a larger research center, hospital, or medical center, its relationships with other departments of the wider facility. In a specialized research center, other than nuclear, maybe merely one or two laboratories are dedicated to work with unsealed sources, while, in a small clinic or an analytical tracer laboratory, only a separate space would be dedicated to it. The siting and layout should also provide for separate personnel and radioactive material entry and exit routes.

Commensurate with the extent and activity of the unsealed sources, the radiation protection program should include all measures to properly delimit, signal and secure all areas where unsealed sources are used, as well as to classify them as controlled areas. If it is the case of a medical facility performing therapy with radiopharmaceuticals, a dedicated ward for patients undergoing such treatments, clearly delimited and signaled, should also be considered, including changing areas and patient toilets and washing facilities.

Some specific areas, such as for source storage and preparation; source sample measurement; and radioactive waste storage and predisposal processing, are to be clearly delimited. The size, type and shielding of these areas, both structural and ancillary, will depend on the amount, energy and activity of the sources. Access to the room, separate space, locked cupboard, safe, refrigerator, or freezer where sources, including generators and radiopharmaceuticals are stored, should be restricted to authorized personnel only; these locations are to be marked with the radiation symbol. Separate storage compartments and an area for the temporary storage of radioactive waste, should also be provided with appropriate protection and signaling.

Wall shielding may be needed to keep a low background in the rooms housing the measuring instruments: radiometers, spectrometers, imaging equipment, etc. Shielding may be also to be considered in the design of a therapy with radiopharmaceuticals division.

The rooms—areas—where unsealed sources are handled, should be preferably isolated from adjacent areas by walls, corridors, benches, or other available structures; or should occupy separate buildings or building wings, if they are radiochemical laboratories involving the use of intermediate to high activity levels. The access to these rooms or areas should be restricted, clearly delimited by lines on the floor and marked with the radiation symbol. To prevent spreading contamination to other areas, the point of access should be provided, at least, with a cloakroom where

**Table 17.3** Radionuclide radiotoxicity groups

| |
|---|
| Group 1: exemption limit < $10^4$ Bq |
| Group 2: exemption limit < $10^5$ Bq |
| Group 3: exemption limit < $10^6$ Bq |
| Group 4: exemption limit < $10^7$ Bq |
| Group 5: exemption limit < $10^8$ Bq |

protective clothing can be stored, put on and taken off; a hand wash-up sink; and a contamination monitor. Larger radiochemical laboratories should be provided with both, clean and unclean point of access, with separate lockers for clothing and personal protective equipment, a corridor with sinks and showers to decontaminate personnel, and fixed monitors to check skin, clothing and equipment contamination.

For low and intermediate activity level laboratories, the maximum activities that can be handled using benches, fume hoods, and glove boxes, or in a given workstation, are typically two or three orders of magnitude higher than established exemption limits [38] and have been frequently associated to the radionuclide radiotoxicity,[4] which is currently classified, for radiation protection purposes, in five risk groups [41, 42], based on the International Safety Standard exemption values [43]. Limits for each group are shown in Table 17.3.

Contained workstations are recommended for easy decontamination. This means that each workstation is provided with its own designated equipment and glassware, including disposable tip automatic pipettes, syringes, and drip trays for minimizing the spread of contamination in the case of spillage; a ventilation system at negative pressure relative to surrounding areas, that includes fume hoods, laminar air flow cabinets, glove boxes, or hot cells (only for higher activities); bricks for shielding; handling devices, including manipulators; containers for solid and liquid radioactive waste collection; sinks for the wash-up of contaminated items, and a sink designated for the authorized discharge of liquid radioactive waste; as well as fixed radiation and contamination monitoring instruments with alarms.

The whole laboratory ventilation system, including the air conditioning system, should be designed such that the airflow should be from areas of minimal likelihood of airborne contamination to areas where such contamination is likely. All air should be vented through the fume hoods and should not be recirculated either directly, in combination with incoming fresh air in a mixing system, or indirectly, as a result of proximity of the exhaust to a fresh air intake.

Drainpipes from sinks should go as directly as possible to the main building sewer and should not connect with other drains within the building, unless those other drains also carry radioactive material. Pipelines through which radioactive materials flow are marked to be monitoring before maintenance. Some laboratories may have a special sewer system terminating in a delay tank.

[4]Radiotoxicity is a measure of how harmful a radionuclide can be to health because of its radioactivity.

All laboratory surfaces should be smooth and nonabsorbent, so that they can be cleaned and decontaminated easily; they include ceilings, walls, benches, bench tops, seats, door, and drawer handles. Laboratory floors should be finished in an impermeable material which is washable and resistant to chemical change, curved to the walls, with all joints sealed and glued to the floor. Bathrooms designated for use within controlled areas should also be finished in materials that are easily decontaminated.

Designation of controlled and supervised areas is paramount to prevent the spread of contamination. Controlled areas are to be drawn in a map of the laboratory, or if the laboratories are part of a larger research center, hospital or medical center, in a map of the nuclear medicine or research facility, indicating its function —storing area, measurement room, sampling and preparation area, contained workstation, point of access, etc.—the radionuclides that may be present, their activity levels and established boundaries. Maps should also indicate fume hoods, laminar air flow cabinets, glove boxes, hot cells (if appropriate), drains, and sinks locations; the limits of supervised areas around controlled areas, if appropriate, and the established points for routine monitoring for both contamination—surface, air—and dose rate monitoring.

As part of the control and accountability of each source, the radiation protection program should consider a semiannual physical inventory of all sources—stored and in use. The inventory should include solid, liquid, and gaseous unsealed sources vials, source preparations and samples, radiopharmaceuticals, radionuclide generators, radiopharmaceutical dispensing equipment, and sealed sources used for calibration or quality control tests. The inventory records should contain the model number of each source, and serial number if one has been assigned, the identity of each source by radionuclide and its nominal activity, the location of each source, and the name of the individual who performed the inventory [38].

The presence and security of each radioactive material in the locations where it is supposed to be stored or permanently used should be checked and recorded on a daily basis. To ensure the continuity in unsealed source control and accountability, it is also very important to have in place writing procedures for its ordering, transportation, receipt, movement within the facility, use, and disposal as radioactive waste, aimed at preventing its theft, loss, and unauthorized withdrawal.

Apart from procedures for the specific uses of the labeled compounds, radiotracers, and radiopharmaceuticals, local rules and procedures should cover provisions: to limit the access to controlled areas to authorized personnel; how and when to wear individual protective equipment; how to minimize occupational radiation exposure during both normal work and unusual events; how to wear, handle and store individual dosimeters; the use of personnel decontamination facilities and authorized decontamination methods; the use of fixed monitors to control hand and glassware decontamination; how and when to conduct a routine workplace monitoring; and cleaning and decontamination procedures, including the method of removing gloves to avoid transferring activity to the hands. Other procedures include measuring instrument use and testing; bioassay samples collection; segregation, collection and disposal of radioactive waste, including waste water from

laundry machines, showers or sinks; etc., along with procedures for compliance supervision, incident reporting, and emergency planning and response (see more possible procedures in the Radiation Protection Program Development Checklist section).

Specific work procedures should be formulated so as to prevent spillage from occurring and, if a spill occurred, to minimize the spread of surface and airborne contamination. Clear rules forbidding eating and drinking, using cosmetics or smoking materials, and operate pipettes by mouth within controlled areas, or bringing crockery or cutlery, cell phones, bags or other personal items into them should also be stated. These procedures should indicate the obligation to always use a drip tray covered with absorbent paper to manipulate or dispense radioactive materials, to cover any cut or break in the skin with a waterproof dressing before entering the controlled area, to mark the glassware designated to each contained workstation, and to ensure an adequate supply of paper towels.

It is recommended to establish specific investigation levels and follow-up actions; for example, an effective dose of 0.5 mSv per month, a finger dose of 15 mSv per month, or a preset value above a historical average to trigger an investigation.

The frequency and content of the personnel training—initial, refresher, on-the-job—including management, administrative and janitorial staff, should be part of the radiation protection program. In addition to the suitability requirements for RSO personnel and authorized users, e.g., a bachelor's or graduate degree from an accredited college or university, a number of years of professional experience, certification from a board whose certification process has been recognized by the NRC or an Agreement State, the training program should include specific initial and refresher training covering subjects such as, basic radionuclide handling techniques; procedures to deal with accidents, spills or contaminated persons, including the proper use of showering and eye washing. It should also include any special instructions and orientation about site specific procedures to offer to fire, police, medical, and other emergency personnel; and to thoroughly prepare site personnel for their responsibilities in the event of postulated accident scenarios. Awareness instruction should be considered for information technology, and janitorial employees.

Individual monitoring is required for any worker who usually works in the controlled area or who occasionally works in it, but is likely to receive an occupational exposure, e.g., maintenance personnel. External doses are assessed using thermoluminescent dosimeters (TLD), film badges, and optical stimulated luminescence dosimeters (OSLD) . If the dosimetry system is provided by the facility, it should be traceable to a secondary standard laboratory; otherwise this service should be delivered by an accredited organization. Internal doses may be evaluated, e.g., by collecting urine samples, assessing the iodine uptake with an external detector, or the quantity of radioactive material with a whole body counter. The committed effective dose should be calculated as part of the worker's total effective dose.

Workplace monitoring includes all routine measurements of dose rate, surface contamination and airborne contamination within the controlled area; special monitoring surveys for specific occasions, activities or tasks, and confirmatory monitoring surveys to check assumptions made about exposure conditions. Dose rate workplace monitoring is typically routinely performed around unsealed source storage locations, at the entrance and midpoint of source and waste storage rooms, in the operator position during different operations with unsealed sources, and in the access points to the controlled area. All working surfaces, sinks, and floors should be weekly routinely monitored using a survey meter or by wipe tests. Contamination of fume hoods, laminar air flow cabinets, glove boxes, and hot cells (if appropriate); ventilation system ducts, filters and traps; and drains, should be monitored before and during maintenance.

Airborne contamination can result from volatiles and aerosols; hence, operations that could generate airborne contamination should only be conducted under fume hoods or glove boxes. Higher activities in any physical form can only be handled in hot cells. Emissions are also reduced using suitable filters, adsorber beds, or scrubbers; which are to be changed regularly and sealed before disposal as radioactive waste. Nonetheless, if airborne contamination is likely, the radiation protection program should consider regular surveys using fixed air samplers, continuous air monitors (CAM), and personal air monitors.

A contamination survey of all working surfaces should be conducted daily before exiting the laboratory. The absence of skin contamination should be confirmed each time after protective gloves have been removed. Protective clothing and shoes are usually monitored when leaving the controlled area. Any item being removed from laboratories should be monitored before departing the controlled area.

Regarding recordkeeping, a copy of the license and its amendments and conditions, as well as any additional certifications or authorizations, should be kept until the license is terminated. Documents of receipt, transfer, and disposal of byproduct materials are to be maintained for as long as the material is possessed, and for 3 years following its transfer or disposal. Records of the semiannual physical inventory, including references to the use of radioactive material logbook and the record of disposal as radioactive waste, should be retained for 3 years. Records of disposal should include date, survey instrument used, and the radiation level measured at the surface of each waste container. Records of spills or other unusual occurrences involving the spread of contamination in and around the facility, equipment, or site, should also be kept until decommissioning; they should include any known information on identification of involved nuclides, quantities, forms, and concentrations. Other records that should be kept for 3 years include personnel training attendance and certification; survey instrument maintenance, and repair; workplace monitoring results, including contamination and airborne monitoring; and portable instrument calibrations, etc.

Certainly, the radiation protection program should also consider its annual review. This annual review is to confirm that the program meets all regulatory requirements, is adequate to ensure the source, workers and public safety, and has

been followed according to the expectations. The review should be also to analyze the program faults and possible improvements. The records of these reviews should also to be kept.

For emergency preparedness, the facility should analyze the possible scenarios of spill of small and large amounts of radioactivity, and the risk of fire or explosion due to the chemicals used and stored. Emergency procedures should include all actions to contain small and large spillages, delimit the affected area and restrict the access, monitor all probable contaminated persons and areas, evacuate all people not involved in the spill, clean the spill, monitor the cleaning operations, collect, label and store the resulting radioactive waste, and decontaminate persons—skin, wounds, and eyes—and the availability and location of emergency protective clothing, decontamination materials, warning notices and barrier tape, portable monitoring instruments, bags and containers, etc. Emergency procedures should also include the allocation of responsibilities, communication and coordination arrangements within the organization and provide for the training of the relevant staff in executing the mitigation measures, which should be periodically rehearsed. If the emergency could lead to a significant release of radioactive material, the arrangements to receive external help by offsite response organizations should also be part of the plan [38].

Laboratories authorized by a specific license for the use of byproduct materials should be formally decommissioned. This means that the program should consider the required steps for the decommissioning of the whole site, separate buildings and/or outdoor areas in case of license termination or operation cessation. The decommissioning should start as soon as practicable after the license is terminated or the operation ceased, so that the mentioned areas are suitable for release. Records of all spills and remaining contamination at the site, separate buildings, or outdoor areas along with information of the radionuclides, quantities, forms, and concentrations involved, should be available at the time of decommissioning, along with a list of all areas designated and formerly designated as controlled areas, including the corresponding as-built drawings and modifications, and locations where current and previous wastes have been buried, if any. The decommissioning program should also include a comprehensive final radiation monitoring to confirm the absence of contamination and any radiation source at the site. Data of the final survey and records of the transfer and disposal of resulting radioactive waste should be submitted to demonstrate that all byproduct material has been properly disposed and that the premises are suitable for release.

## 17.7 Radiation Protection Program Development Checklist

The set of questions listed in Table 17.4 below, although not completely exhaustive, can help when developing a radiation protection program.

**Table 17.4** Radiation protection program questionary

| *Responsibility and accountability for safety* | | |
|---|---|---|
| 1 | Does the radiation protection program start with a statement outlining the safety policy and accountability for radiation safety? | ☐ Yes ☐ No |
| 2 | Is the radiation protection program approved and signed by the senior management? | ☐ Yes ☐ No |
| 3 | Is the radiation protection program reviewed on a periodic basis? | ☐ Yes ☐ No |
| 4 | Does the radiation protection program have the goals for the specific period clearly specified? | ☐ Yes ☐ No |
| 5 | Does the radiation protection program include criteria for evaluating its performance, e.g., administrative limits, constraints, investigation, and action levels? | ☐ Yes ☐ No |
| 6 | Is the scope of the radiation protection program adequate, e.g., does it address all potential hazards to which workers are exposed at the facility? | ☐ Yes ☐ No |
| 7 | Has the radiation protection program also outlined how it is implemented at the facility and who is responsible for what? | ☐ Yes ☐ No |
| 8 | Does the radiation protection program state its limitations? | ☐ Yes ☐ No |
| 9 | Does it clearly state that all departments, groups, or individuals in the organization using radiation sources and/or radioactive materials are responsible for complying with the radiation protection program? | ☐ Yes ☐ No |
| 10 | Are the responsibilities and duties of these departments, groups, or individuals regarding the radiation protection program clearly stated? | ☐ Yes ☐ No |
| 11 | Does the facility have a radiation safety officer (RSO)? | ☐ Yes ☐ No |
| 12 | Does the facility have an investigation level approved at management level for an effective dose value at or above which an investigation should be conducted? | ☐ Yes ☐ No |
| 13 | Does the facility have an investigation level approved at management level for an intake value at or above which an investigation should be conducted? | ☐ Yes ☐ No |
| 14 | Does the facility have an investigation level approved at management level for a contamination value per unit area at or above which an investigation should be conducted? | ☐ Yes ☐ No |
| 15 | Are the results of individual and workplace monitoring effectively analyzed and informed to senior management at least once a year? | ☐ Yes ☐ No |
| 16 | Does the facility have local radiation safety rules developed and implemented, including applicable operational radiation protection procedures and supervision of compliance procedures? | ☐ Yes ☐ No |
| 17 | Are the local radiation safety rules approved by the senior management? | ☐ Yes ☐ No |
| 18 | Are the local radiation safety rules reviewed and updated on a regular basis? | ☐ Yes ☐ No |
| 19 | Are the local safety rules and procedures adequately governing the use of radiation source at the facility? | ☐ Yes ☐ No |
| 20 | Does the radiation safety officer regularly supervise compliance with local safety rules and procedures? | ☐ Yes ☐ No |
| 21 | Is all the relevant information related to licenses, registrations, limits, and technical conditions for source operation properly documented? | ☐ Yes ☐ No |
| 22 | Is all the relevant information related to radiation source specifications and calibrations properly registered and documented? | ☐ Yes ☐ No |

(continued)

**Table 17.4** (continued)

| | | |
|---|---|---|
| 23 | Is all the relevant information related to radiation sources and radioactive materials transfer registered and documented? | ☐ Yes ☐ No |
| 24 | Are all relevant reports regarding assessment of the radiological situation at the facility properly filed? | ☐ Yes ☐ No |
| 25 | Is all the relevant information regarding radiation protection and safety decisions registered and documented? | ☐ Yes ☐ No |
| 26 | Does the facility annually review the radiation protection program based on the compliance report and the radiological situation at the facility? | ☐ Yes ☐ No |
| 27 | Does the facility regularly review that radiation safety activities are performed in accordance with licensee-approved procedures and regulatory requirements? | ☐ Yes ☐ No |
| 28 | Does the radiation safety officer (committee) advise the senior management on technical and regulatory issues regarding the radiation safety program? | ☐ Yes ☐ No |
| 29 | Does the radiation safety officer (committee) take part in the planning of activities involving significant exposures, and advice on the conditions under which work can be undertaken? | ☐ Yes ☐ No |
| 30 | Does the radiation safety officer initiate, recommend, or provide corrective actions in case of radiation incidents? | ☐ Yes ☐ No |
| 31 | Does the radiation safety officer initiate, recommend, or provide corrective actions in case of issues with or violations of the local radiation safety rules and procedures? | ☐ Yes ☐ No |
| 32 | Does the radiation safety officer stop, when necessary, unsafe operations? | ☐ Yes ☐ No |
| 33 | Does the radiation safety officer periodically organize inspections of the controlled areas? | ☐ Yes ☐ No |
| 34 | Has the radiation safety officer (committee) regularly performed radiation safety optimization analysis of specific activities or groups, and initiate actions to reasonably reduce dose? | ☐ Yes ☐ No |
| 35 | Does the radiation safety officer inform the management on the findings of controlled area inspections after completed? | ☐ Yes ☐ No |
| 36 | Does the senior management, together with the radiation safety officer (committee), verify the implementation of corrective actions? | ☐ Yes ☐ No |
| 37 | Does the senior management, together with the radiation safety officer (committee), review and approve the uses of radioactive material within the facility? | ☐ Yes ☐ No |
| 38 | Does the senior management organize periodic audits of the radiation protection program? | ☐ Yes ☐ No |
| 39 | Are corrective actions documented? | ☐ Yes ☐ No |
| 40 | Are inspection results documented? | ☐ Yes ☐ No |
| 41 | Are audit results documented? | ☐ Yes ☐ No |
| 42 | Is all the relevant information related to effluent release and radioactive waste disposal, properly registered and documented? | ☐ Yes ☐ No |
| 43 | Does the facility have an inventory of radiation sources and equipment containing radiation sources? | ☐ Yes ☐ No |
| 44 | Does the facility have an inventory of unsealed radioactive sources? | ☐ Yes ☐ No |
| 45 | Are there procedures in place to maintain and update the source inventories? | ☐ Yes ☐ No |

(continued)

**Table 17.4** (continued)

| | | |
|---|---|---|
| 46 | Are the inventories maintained and updated in line with the provisions and procedures? | ☐ Yes ☐ No |
| 47 | Does the radiation safety officer regularly (at least once a year) check out the inventories? | ☐ Yes ☐ No |
| 48 | Is the management regularly informed on the source inventories status? | ☐ Yes ☐ No |
| *Personnel suitability* | | |
| 49 | Are the skills or qualifications required to perform the responsibilities and duties of the radiation protection program adequately stated? | ☐ Yes ☐ No |
| 50 | It is ensured that the skills and qualification requirements are met when hiring the personnel? | ☐ Yes ☐ No |
| 51 | Does the facility provide training to effectively meet the responsibilities and duties of the radiation protection program? | ☐ Yes ☐ No |
| 52 | Does the facility ensure training to all personnel, including management and administrative staff? | ☐ Yes ☐ No |
| 53 | Is all the relevant information related to the content, duration, and frequency of education and training programs, satisfactorily documented? | ☐ Yes ☐ No |
| 54 | Are refresher courses provided on a periodic basis? | ☐ Yes ☐ No |
| 55 | Does the facility schedule safety and hygiene workplace assessments by occupational medicine physicians on a regular basis? | ☐ Yes ☐ No |
| 56 | Does the facility schedule medical examinations by occupational medicine physicians after an abnormal exposure of workers? | ☐ Yes ☐ No |
| 57 | Does the facility schedule medical examinations by occupational medicine physicians before the worker employment terminates? | ☐ Yes ☐ No |
| *Technical adequacy (engineered controls)* | | |
| 58 | Does the facility have adequate containment devices and systems to control the sources of contamination? | ☐ Yes ☐ No |
| 59 | Does the facility have adequate methods, systems, and devices to control access to potential exposure areas? | ☐ Yes ☐ No |
| 60 | Does the facility have adequate radiation shielding structures and buildings to reduce radiation fields to acceptable levels? | ☐ Yes ☐ No |
| 61 | Do workplace areas have sufficient lead bricks and plastic panels for local shielding? | ☐ Yes ☐ No |
| 62 | Does the facility have adequate ventilation systems in all buildings where surface or airborne contamination may be present? | ☐ Yes ☐ No |
| 63 | Do all areas where surface or airborne contamination may be present have enough local exhaust systems? | ☐ Yes ☐ No |
| 64 | Are exhaust systems equipped with HEPA filters? | ☐ Yes ☐ No |
| 65 | Are exhaust systems equipped with activated charcoal filters? | ☐ Yes ☐ No |
| 66 | Does the facility have adequate methods and capabilities for the decontamination of equipment and tools? | ☐ Yes ☐ No |
| 67 | Does the facility have adequate methods and capabilities for the decontamination of protective clothing? | ☐ Yes ☐ No |
| 68 | Does the facility have adequate methods and capabilities for the decontamination of personnel? | ☐ Yes ☐ No |

(continued)

**Table 17.4** (continued)

| | | |
|---|---|---|
| 69 | Does the facility have adequate change rooms, interlocks, or delimiting zones to control access to potentially contaminated areas? | ☐ Yes ☐ No |
| 70 | Does the facility have adequate means and vehicles for the transportation of contaminated equipment? | ☐ Yes ☐ No |
| 71 | Does the facility have an adequate storehouse for contaminated equipment? | ☐ Yes ☐ No |
| 72 | Does the facility have appropriate storerooms for the radiation sources when they are not in use? | ☐ Yes ☐ No |
| 73 | Does the facility have an appropriate storeroom—cupboard, closet, cabinet, etc.—for storing unsealed radioactive sources? | ☐ Yes ☐ No |
| 74 | Do the store locations have any dedicated fume hood or glove box, and refrigerator as needed? | ☐ Yes ☐ No |
| 75 | Are all radioactive source stores isolated from other areas, correctly delimited, guarded, and signaled? | ☐ Yes ☐ No |
| 76 | Are all source storerooms adequately locked with a key, padlock, combination lock, etc.? | ☐ Yes ☐ No |
| 77 | Are unsealed source store locations adequately locked with a key, padlock, combination lock, etc.? | ☐ Yes ☐ No |
| 78 | Is the radiation source storage documentation updated on a routine basis? | ☐ Yes ☐ No |
| 79 | Is the unsealed radioactive source storage documentation updated in keeping with the source using? | ☐ Yes ☐ No |
| 80 | Does the facility have provisions for the reception, storage, inventory, and disposal of radioactive materials and radiation sources? | ☐ Yes ☐ No |
| 81 | Are the workplaces, or laboratories, for the use of unsealed source equipped with enough fume hoods and glove boxes? | ☐ Yes ☐ No |
| 82 | Does the facility use a proper area classification in the sites, where the work with radiation sources and radioactive materials is performed? | ☐ Yes ☐ No |
| 83 | Is the area classification regularly reviewed as appropriate? | ☐ Yes ☐ No |
| 84 | Are the classified areas properly designated, delimited, and signaled? | ☐ Yes ☐ No |
| 85 | Does the facility keep an updated map, or record, documenting the designation and location of controlled and supervised areas? | ☐ Yes ☐ No |
| *Surveillance* | | |
| 86 | Does the facility have an individual monitoring program for external exposure of workers? | ☐ Yes ☐ No |
| 87 | Does the facility have an individual monitoring program for internal exposure of workers? | ☐ Yes ☐ No |
| 88 | Does the facility obtain approved personal dosimetry services from commercial suppliers? | ☐ Yes ☐ No |
| 89 | Does the facility have procedures to rule the personal radiation monitoring, i.e., dosimeter wearing, collection, reading, calculation, recording, etc.? | ☐ Yes ☐ No |
| 90 | Is the individual monitoring performed in the indicated frequency? | ☐ Yes ☐ No |
| 91 | Do the workers wear dosimeters in various parts of the body, e.g., lens of the eye, extremities, etc., during work? | ☐ Yes ☐ No |
| 92 | Do the workers wear an additional direct reading dosimeter and/or a warning device during work? | ☐ Yes ☐ No |

(continued)

**Table 17.4** (continued)

| | | |
|---|---|---|
| 93 | Is it likely that the committed effective dose from annual intakes of radionuclides in the facility exceeds 1 mSv? | ☐ Yes ☐ No |
| 94 | If it is likely that the committed effective dose exceeds 1 mSv, does the facility obtain internal dosimetry services, e.g., whole body counter or specific organ counter from commercial suppliers? | ☐ Yes ☐ No |
| 95 | Does the facility have its own whole body counter for internal dosimetry? | ☐ Yes ☐ No |
| 96 | Does the facility have its own specific organ counter for internal dosimetry? | ☐ Yes ☐ No |
| 97 | Does the facility have its own bioassay methods and measurements to monitor intakes? | ☐ Yes ☐ No |
| 98 | Does the facility measure the activity in physical samples, such as filters from personal or fixed air samplers? | ☐ Yes ☐ No |
| 99 | Does the facility monitor the intakes on a routine basis? | ☐ Yes ☐ No |
| 100 | Does the facility monitor the intakes on a task related basis? | ☐ Yes ☐ No |
| 101 | Does the facility monitor the intakes on a special basis? | ☐ Yes ☐ No |
| 102 | Does the facility have procedures ruling internal dosimetry, i.e., frequency, sample collection, calculation, recording, etc.? | ☐ Yes ☐ No |
| 103 | Are individual monitoring results (external and internal) documented? | ☐ Yes ☐ No |
| 104 | Does each worker have its personal dose record? | ☐ Yes ☐ No |
| 105 | Are personal dose records effectively maintained and updated? | ☐ Yes ☐ No |
| 106 | Does the facility have a workplace monitoring program implemented? | ☐ Yes ☐ No |
| 107 | Does the facility conduct a workplace monitoring in all working areas within the workplace on a routine basis? | ☐ Yes ☐ No |
| 108 | Does the facility conduct a workplace monitoring in specific working areas within the workplace on a special basis? | ☐ Yes ☐ No |
| 109 | Does the facility conduct task related workplace monitoring in specific areas within the workplace? | ☐ Yes ☐ No |
| 110 | Does the facility conduct a comprehensive radiological survey when any new installation is put into service? | ☐ Yes ☐ No |
| 111 | Does the facility conduct a comprehensive radiological survey when any substantial changes have been made in an existing installation? | ☐ Yes ☐ No |
| 112 | Are there preestablished points for workplace monitoring clearly identified? | ☐ Yes ☐ No |
| 113 | Are there preestablished points for workplace monitoring for the specific supervised area clearly represented on the map or record? | ☐ Yes ☐ No |
| 114 | Is routine workplace monitoring performed in the indicated frequency? | ☐ Yes ☐ No |
| 115 | Are before-work measurements performed in preestablished points consistent with established radiation protection procedures? | ☐ Yes ☐ No |
| 116 | Are during-work measurements performed in preestablished points consistent with established radiation protection procedures? | ☐ Yes ☐ No |
| 117 | Are after-work measurements performed in preestablished points consistent with established radiation protection procedures? | ☐ Yes ☐ No |
| 118 | Is workplace monitoring performed consistent with established radiation protection procedures? | ☐ Yes ☐ No |
| 119 | Are the radiation protection procedures for workplace monitoring reviewed on a periodic basis? | ☐ Yes ☐ No |

(continued)

**Table 17.4** (continued)

| | | |
|---|---|---|
| 120 | Are there permanently installed monitoring systems, or devices, in locations where an unexpected increase of the radiation level might occur? | ☐ Yes ☐ No |
| 121 | Are fixed monitors fitted with appropriate audio, and/or visual alarms, to warn of unacceptable conditions? | ☐ Yes ☐ No |
| 122 | Does the facility have adequate portable instruments for measuring workplace surface contamination? | ☐ Yes ☐ No |
| 123 | Does the facility have adequate laboratories to measure and analyze wipe samples from surface contamination? | ☐ Yes ☐ No |
| 124 | Does the facility have adequate laboratories for measuring and analyzing filter samples from air contamination? | ☐ Yes ☐ No |
| 125 | Does the facility have the adequate quantity of portable instruments for measuring the workplace surface contamination? | ☐ Yes ☐ No |
| 126 | Does the facility have the adequate quantity of portable instruments for measuring the workplace dose rate? | ☐ Yes ☐ No |
| 127 | Do the portable instruments have the adequate range? | ☐ Yes ☐ No |
| 128 | Do the monitors have the adequate range? | ☐ Yes ☐ No |
| 129 | Does the facility obtain the repair of defective instruments and monitors from a technical supplier workshop? | ☐ Yes ☐ No |
| 130 | Does the facility obtain maintenance services from an external supplier? | ☐ Yes ☐ No |
| 131 | Does the facility have a proper workshop for the repair of defective portable instruments and fixed monitors? | ☐ Yes ☐ No |
| 132 | Does the facility have a proper workshop for instrument and measuring devices maintenance? | ☐ Yes ☐ No |
| 133 | Is the instrument maintenance and repair documented? | ☐ Yes ☐ No |
| 134 | Do the portable instruments have the adequate sensitivity? | ☐ Yes ☐ No |
| 135 | Is the conformance testing of workplace monitoring instruments performed before the first use? | ☐ Yes ☐ No |
| 136 | Does the facility only rely upon the specifications provided by the manufacturer? | ☐ Yes ☐ No |
| 137 | Is the periodic or calibration testing of workplace monitoring instruments carried out at least once a year? | ☐ Yes ☐ No |
| 138 | Is the performance testing of workplace monitoring instruments carried out by an authorized organization at least every 2–3 years? | ☐ Yes ☐ No |
| 139 | Does the facility have a calibration bench to perform the portable monitoring instruments periodic calibration testing? | ☐ Yes ☐ No |
| 140 | Does the facility have secondary calibration standards? | ☐ Yes ☐ No |
| 141 | Are the secondary calibration standards measured against primary calibration standards with the established frequency? | ☐ Yes ☐ No |
| 142 | Is the calibration of workplace monitoring instruments performed with standards traceable to national standards? | ☐ Yes ☐ No |
| 143 | Is the calibration performed against established standards? | ☐ Yes ☐ No |
| 144 | Does the facility obtain calibration services from a commercial authorized organization? | ☐ Yes ☐ No |
| 145 | Is the instrument calibration documented? | ☐ Yes ☐ No |

(continued)

**Table 17.4** (continued)

| | | |
|---|---|---|
| 146 | Does the equipment calibration record include the calibration accuracy over the range of operation for the type of radiation to monitor, the date of the test, identification of the calibration standards used, calibration frequency, and the name and signature of the qualified person under whose direction the test was carried out? | ☐ Yes ☐ No |
| 147 | Do portable monitoring instruments have adequate accuracy? | ☐ Yes ☐ No |
| 148 | Do portable monitoring instruments have adequate precision? | ☐ Yes ☐ No |
| 149 | Are the uncertainties of monitoring instruments documented? | ☐ Yes ☐ No |
| 150 | Are the workplace monitoring instruments frequently source-checked before use to ensure their proper functioning? | ☐ Yes ☐ No |
| 151 | Are measurement procedures approved at the appropriate level of management? | ☐ Yes ☐ No |
| 152 | Are measurement procedures developed by the Radiation Safety (EHS or ISHN) department? | ☐ Yes ☐ No |
| 153 | Are measurement procedures developed by a specialized department, e.g., dosimetry, radiometry, or metrology department? | ☐ Yes ☐ No |
| 154 | Are measurement procedures governing the use of laboratory equipment appropriate? | ☐ Yes ☐ No |
| 155 | Are measurement procedures governing the use of portable instruments appropriate? | ☐ Yes ☐ No |
| 156 | Are the results of workplace monitoring documented appropriately? | ☐ Yes ☐ No |
| 157 | Do records of workplace monitoring include the instrument, date, time, location, and name of the person who carried out the measurement? | ☐ Yes ☐ No |
| *Emergency planning and preparedness* | | |
| 158 | Does the facility have an emergency plan and emergency procedures to respond to potential incidents? | ☐ Yes ☐ No |
| 159 | Is the emergency plan and emergency procedures approved by the higher management? | ☐ Yes ☐ No |
| 160 | Does the facility periodically train its personnel for these emergency procedures? | ☐ Yes ☐ No |
| 161 | Does the facility organize games and drills to exercise its personnel for specific emergency measures at least once a year? | ☐ Yes ☐ No |
| 162 | Is all relevant information related to incidents and accidents, properly registered and documented? | ☐ Yes ☐ No |
| *Radioactive waste* | | |
| 163 | Does the facility generate radioactive waste? | ☐ Yes ☐ No |
| 164 | Does the facility have specific procedures for classification and segregation of radioactive waste? | ☐ Yes ☐ No |
| 165 | Are these procedures reviewed and updated on a regular basis? | ☐ Yes ☐ No |
| 166 | Is solid radioactive waste collected in yellow colored plastic bags, and placed in covered containers, trash bins, or drums? | ☐ Yes ☐ No |
| 167 | Are solid radioactive waste containers, trash bins, or drums, labeled with the radiation symbol and tagged with necessary data to identify the waste? | ☐ Yes ☐ No |
| 168 | Are solid radioactive waste containers, trash bins, or drums, kept apart from common waste containers? | ☐ Yes ☐ No |

(continued)

**Table 17.4** (continued)

| | | |
|---|---|---|
| 169 | If the facility generates sharp waste, are sharps collected aside in sturdy approved containers? | ☐ Yes ☐ No |
| 170 | If the facility generates infected solid waste, is infected solid radioactive waste collected aside in red colored plastic bags placed in covered containers, trash bins, or drums? | ☐ Yes ☐ No |
| 171 | Are infected solid radioactive waste containers, trash bins or drums additionally labeled with the biohazard symbol and tagged with necessary data to identify the waste? | ☐ Yes ☐ No |
| 172 | Are empty lead pigs and scintillation vials collected aside in plastic lined containers? | ☐ Yes ☐ No |
| 173 | Does the facility generate biological solid waste—animal carcasses, excretes, organ, tissues, and beddings? | ☐ Yes ☐ No |
| 174 | Is the biological solid waste double bagged in red colored bags, properly labeled, and frozen? | ☐ Yes ☐ No |
| 175 | Is the biological solid waste wrapped in bench blankets, and/or bags filled with sorbent materials, to avoid moisture seeping through the bag while the waste is in storage or transit? | ☐ Yes ☐ No |
| 176 | Is aqueous liquid radioactive waste separately collected in a plastic or chemically compatible covered jug or carboy, labeled with the radiation symbol, and placed inside a secondary containment? | ☐ Yes ☐ No |
| 177 | Is the secondary container for aqueous liquid waste large enough to hold all the liquid in the primary container, plus 10 %, and prevent spills or leaks? | ☐ Yes ☐ No |
| 178 | Is the secondary container for aqueous liquid waste filled with an absorbent for the liquid waste? | ☐ Yes ☐ No |
| 179 | Is organic liquid radioactive waste separately collected in a plastic or chemically compatible covered jug or carboy, labeled with the radiation symbol, and placed inside a secondary containment? | ☐ Yes ☐ No |
| 180 | Does the organic liquid radioactive waste pose flammable, corrosive, toxic, explosive, or reactive hazard? | ☐ Yes ☐ No |
| 181 | Is organic liquid radioactive waste—liquid scintillation cocktails, organic solvents, toxic metals, etc.—collected aside? | ☐ Yes ☐ No |
| 182 | Does the organic liquid waste contain iodine radionuclides? | ☐ Yes ☐ No |
| 183 | Does the facility treat the organic liquid radioactive waste by incineration? | ☐ Yes ☐ No |
| 184 | Does the facility treat the organic liquid radioactive waste by distillation to recover solvents? | ☐ Yes ☐ No |
| 185 | Does the facility incinerate the biological solid waste? | ☐ Yes ☐ No |
| 186 | Does the facility have records of the incinerated waste? | ☐ Yes ☐ No |
| 187 | Does the facility have records of the distilled waste? | ☐ Yes ☐ No |
| 188 | Does the facility discharge any aqueous liquid waste directly down the drain? | ☐ Yes ☐ No |
| 189 | Does the liquid aqueous waste discharged directly down the drain comply with the established sewerable limits stated by 10 CFR Part 20? | ☐ Yes ☐ No |
| 190 | Does the facility have a sink selected and labeled for liquid direct discharge purpose? | ☐ Yes ☐ No |
| 191 | Does the facility have records of area surveying for the sink and its surroundings? | ☐ Yes ☐ No |

(continued)

**Table 17.4** (continued)

| | | |
|---|---|---|
| 192 | Does the facility have records of the releases made through the sink? | ☐ Yes ☐ No |
| 193 | Do these records state the date, quantity, and radionuclide concentration of the aqueous liquid waste released? | ☐ Yes ☐ No |
| 194 | Radioactive waste is also segregated according to their half-life in: (a) 10 h or less; (b) less than 10 days; and, (c) less than 100 days? | ☐ Yes ☐ No |
| 195 | Does the facility have a temporary store for radioactive waste decay-in-storage? | ☐ Yes ☐ No |
| 196 | Does the temporary storage have sufficient room to handle, classify, and accommodate the waste for as long as it decays? | ☐ Yes ☐ No |
| 197 | Is the store shielded to comply with those requirements? | ☐ Yes ☐ No |
| 198 | Does the facility have records of the storage radiological surveys? | ☐ Yes ☐ No |
| 199 | Does the facility have records of the waste stored? | ☐ Yes ☐ No |
| 200 | Does the facility have records of the waste discharge from storage? | ☐ Yes ☐ No |
| 201 | Does the facility have records regarding the systematic sampling and measurement of stored waste? | ☐ Yes ☐ No |
| 202 | Does the facility regularly transfer the solid radioactive waste to a waste operator for processing and disposal? | ☐ Yes ☐ No |
| 203 | Is collected bagged waste then packaged in sturdy fiber drums or cardboard boxes lined with plastic, sealed, and labeled for temporary storage on-site? | ☐ Yes ☐ No |
| 204 | Is the solid waste packaged in compliance with waste operator waste acceptance criteria (WAC)? | ☐ Yes ☐ No |
| 205 | Does the facility regularly transfer the biological solid radioactive waste to a waste operator for processing and disposal? | ☐ Yes ☐ No |
| 206 | Is biological solid waste packaged in compliance with waste operator waste acceptance criteria (WAC)? | ☐ Yes ☐ No |
| 207 | Does the facility transfer the liquid radioactive waste to a waste operator for processing and disposal? | ☐ Yes ☐ No |
| 208 | Is collected liquid waste then chemically adjusted and packaged in compliance with the waste operator waste acceptance criteria (WAC)? | ☐ Yes ☐ No |
| 209 | Does the facility have records of the waste transferred to the waste operator for processing and disposal? | ☐ Yes ☐ No |
| 210 | Does the facility have a signed agreement with the supplier to return the radiation sources after their use-life? | ☐ Yes ☐ No |
| 211 | Does the facility have a temporary storage for the on-site decay of the spent or disused sources with a half-life of 120 days or less? | ☐ Yes ☐ No |
| 212 | Does the facility have records of the sources disposed of after reaching the clearance level? | ☐ Yes ☐ No |
| 213 | Does the facility transfer the spent or disused sources to a waste operator for processing and disposal? | ☐ Yes ☐ No |
| 214 | Does the facility have records of the spent or disused sources transferred to the waste operator for processing and disposal? | ☐ Yes ☐ No |
| *Decommissioning* | | |
| 215 | Does the facility have a program for the formal decommissioning? | ☐ Yes ☐ No |
| 216 | Does it include a previously signed arrangement with the manufacturer or distributor for the return of the radiation sources? | ☐ Yes ☐ No |

(continued)

**Table 17.4** (continued)

| | | |
|---|---|---|
| 217 | Does it include the transfer of any radiation source to other organization? | ☐ Yes ☐ No |
| 218 | Does it include the transfer of containers made of depleted uranium (DU) to other organization? | ☐ Yes ☐ No |
| 219 | Does it include the transfer of *X*-ray generators to another organization? | ☐ Yes ☐ No |
| 220 | Does it include the transfer of neutron generators to another organization? | ☐ Yes ☐ No |
| 221 | Does it include the transfer of any radiation source to a disposal operator to be disposed of? | ☐ Yes ☐ No |
| 222 | Does it include the transfer of containers made of depleted uranium (DU) to a disposal operator to be disposed of? | ☐ Yes ☐ No |
| 223 | Does it include the transfer of neutron generator target sources to a disposal operator to be disposed of? | ☐ Yes ☐ No |
| 224 | Does it include the transfer of containers made of depleted uranium (DU) to the U.S. Department of Energy (DOE)? | ☐ Yes ☐ No |
| 225 | Does it include making inoperable all *X*-ray generators before being disposed of? | ☐ Yes ☐ No |
| 226 | Does it include making inoperable all neutron generators before being transferred to a disposal operator to be disposed of? | ☐ Yes ☐ No |
| 227 | Does it include the transfer and disposal of radioactive waste resulting from laboratory decommissioning operations? | ☐ Yes ☐ No |
| 228 | Is the receiving organization authorized to possess and use the radiation source to be transferred and have the corresponding license? | ☐ Yes ☐ No |
| 229 | Have the transfer formalities been done through the NRC or Agreement State? | ☐ Yes ☐ No |
| 230 | Will the transfer documents include the results of a leak test, and an activity certification emitted by an organization specifically authorized by the NRC or an Agreement State? | ☐ Yes ☐ No |
| 231 | Are the facility keeping records of any spills occurred and the remaining contamination at the site, separate buildings or outdoor areas, along with information of the radionuclides, quantities, forms, and concentrations involved? | ☐ Yes ☐ No |
| 232 | Does the facility have a list of all areas designated and formerly designated as controlled areas? | ☐ Yes ☐ No |
| 233 | Does the facility have the designated controlled area as-built drawings and further modifications made to them? | ☐ Yes ☐ No |
| 234 | Have the facility well identified the locations where radioactive waste has been buried? | ☐ Yes ☐ No |
| 235 | Is the facility keeping records of any radioactive waste burial? | ☐ Yes ☐ No |
| 236 | Does the decommissioning program include a comprehensive final radiation monitoring to confirm the absence of contamination and any radiation source at the site? | ☐ Yes ☐ No |
| 237 | Does the program include the storage of the sources, generators, and other radioactive materials, in a safe condition until packaged and transported to another organization, disposal operator, or original manufacturer or distributor? | ☐ Yes ☐ No |
| 238 | Does the facility foresee the construction of a temporary storage for the resulting radioactive waste? | ☐ Yes ☐ No |

## References


1. U.S. NRC (2015) 10 CFR Part 20—Standards for protection against radiation. United States Nuclear Regulatory Commission. http://www.nrc.gov/reading-rm/doc-collections/cfr/part020/
2. U.S. NRC (2015) Part 36. Licenses and radiation safety requirements for irradiators. United States Nuclear Regulatory Commission. http://www.nrc.gov/reading-rm/doc-collections/cfr/part036/full-text.html
3. Vacca PC, Collins DJ, Mitchell MW, Radcliffe WH, Schwartz ME (1998) Consolidated guidance about materials licenses program-specific guidance about self-shielded irradiator licenses. NUREG-1556, vol 5. Washington, U.S, Nuclear Regulatory Commission
4. Conference of Radiation Control Program Directors (2015) A partnership dedicated to radiation protection, "SSRCR's," Conference of radiation control program directors, Inc. http://crcpd.org/SSRCRs/default.aspx
5. International Atomic Energy Agency (2010) Radiation safety of gamma, electron and X ray irradiation facilities. Safety Standards Series No. SSG-8. IAEA, Vienna
6. International Atomic Energy Agency (2008) Trends in radiation sterilization of health care products STI/PUB/1313. IAEA, Vienna
7. International Atomic Energy (2005) Categorization of radioactive sources. Safety Guide No. RS-G-1.9. IAEA, Vienna
8. GE Inspection Technologies (2007) Industrial radiography. image forming techniques GEIT-30158 (01/07), General Electric Company
9. UNSCEAR 2008 (2011) Source and effects of ionizing radiation, Report to the general assembly with scientific annexes, vol 2, Scientific Annex C: Radiation Exposures in Accidents," UN, New York
10. U.S. NRC (1997) 10 CFR Part 34 licenses for industrial radiography and radiation safety requirements for industrial radiographic operations. United States Nuclear Regulatory Commission, 1997, last amended 2015. http://www.nrc.gov/reading-rm/doc-collections/cfr/part034/
11. Conference of Radiation Control Program Directors (2015) Industrial Radiography Certification, Conference of Radiation Control Program Directors. A partnership dedicated to radiation protection. http://www.crcpd.org/IR.aspx
12. International Atomic Energy Agency (2011) Radiation safety in industrial radiography. IAEA Specific Safety Guide No. SSG-11. IAEA, Vienna
13. International Standard ISO 3999 Second edition (2004) Radiation protection—Apparatus for industrial gamma radiography—Specifications for performance, design and tests ISO 3999:2004(E) 2004. ISO, Geneva
14. 49 CFR, Part 172 (1998) Hazardous Materials Regulations, U.S. Department of Transportation
15. International Atomic Energy Agency (2012) Regulations for the safe transport of radioactive material 2012 edition. IAEA Safety Standards Series No. SSR-6. IAEA, Vienna
16. U.S. Nuclear Regulatory Commission (1997) Consolidated guidance about materials licenses. Program-specific guidance about portable gauge licenses. NUREG-1556, vol. 1. U.S. Nuclear Regulatory Commission, Washington
17. U.S. Nuclear Regulatory Commission (1998) Consolidated guidance about materials licenses. Program-specific guidance about fixed gauge licenses. NUREG-1556, vol. 4. U.S. Nuclear Regulatory Commission, Washington
18. U.S. NRC (2015) Specific licenses of broad scope. United States Nuclear Regulatory Commission. http://www.nrc.gov/reading-rm/doc-collections/cfr/part033/part033-0011.html
19. U. S. Nuclear Regulatory Commission (2000) A risk analysis of fixed nuclear gauges. NUREG-1669. U.S. Nuclear Regulatory Commission, Washington
20. International Organization for Standardization (ISO) (1992) Radiation protection—Sealed radioactive sources—Leakage test methods ISO 9978: 1992(E). ISO, Geneva
21. International Atomic Energy Agency (2012) Neutron generators for analytical purposes. IAEA Radiation Technology Reports Series No. 1. IAEA, Vienna

22. APNGA (2016) Portable nuclear gauge training center for Troxler, CPN, Humboldt, Seaman & InstroTek Gauges. American Portable Nuclear Gauge Association. https://www.apnga.com/
23. American Technical Institute (ATI) (2015) Nuclear gauge training. American Technical Institute, LLC. http://atiinspectortraining.com/index.php/nuclear-gauge-training/
24. 49 CFR, Part 172, Subpart H (1998) Hazardous materials regulations. Training, U.S. Department of Transportation
25. International Atomic Energy Agency (2003) Radiation protection and the management of radioactive waste in the oil and gas industry. Safety Reports Series No.34. IAEA, Vienna
26. U.S. Nuclear Regulatory Commission (2014) Consolidated guidance about materials licenses. Program-specific guidance about well logging, tracer, and field flood study licenses. NUREG-1556, vol 14, Rev. 1. United States Nuclear Regulatory Commission, Washington
27. U.S. NRC (1987) Part 39—Licenses and Radiation Safety Requirements for well logging. United States Nuclear Regulatory Commission. http://www.nrc.gov/reading-rm/doc-collections/cfr/part039/full-text.html
28. U. S. Nuclear Regulatory Commission (2000) Consolidated guidance about materials licenses: program-specific guidance about well logging, tracer, and field flood study licenses. NUREG-1556, vol 14. U.S. Nuclear Regulatory Commission, Washington
29. U.S. Nuclear Regulatory Commission (2000) Risk analysis and evaluation of regulatory options for nuclear byproduct material systems. NUREG/CR-6642, United States Nuclear Regulatory Commission, Washington
30. International Atomic Energy Agency (2010) Radiation protection and the management of radioactive waste in the oil and gas industry. TCS No. 40. IAEA, Vienna
31. QSA Global (2007) Sentinel Model 650L source changer operating instructions MAN-020, QSA Global Inc
32. U.S. Customs and Border Protection (2015) Technology CBP uses for enforcement—Vehicle and cargo inspection system (VACIS). U.S. Customs and Border Protection. https://help.cbp.gov/app/answers/detail/a_id/317/～/technology-cbp-uses-for-enforcement—vehicle-and-cargo-inspection-system
33. ICRP Publication 125 (2014) Radiological protection in security screening. Ann ICRP 43(2)
34. United States Interagency Steering Committee on Radiation Standards (2008) Guidance for security screening of humans utilizing ionizing radiation. ISCORS, Washington
35. NCRP Commentary No.16 (2003) Screening of humans for security purposes using ionizing radiation scanning systems. National Council on Radiation Protection and Measurements, Bethesda
36. American National Standard ANSI/HPS N43.17-2009 (2009) Radiation safety for personnel security screening systems using x-ray or gamma radiation. Health Physics Society, McLean
37. World Customs Organization (WCO) (2011) Guidelines for the procurement and deployment of scanning/NII equipment. World Customs Organization, Brussels
38. U.S. NRC (2016) Part 30-Rules of general applicability to domestic licensing of byproduct material. United States Nuclear Regulatory Commission. http://www.nrc.gov/reading-rm/doc-collections/cfr/part030/
39. U.S. NRC (2016) Part 33—Specific domestic licenses of broad scope for byproduct material. United States Nuclear Regulatory Commission. http://www.nrc.gov/reading-rm/doc-collections/cfr/part033/
40. U.S. NRC (2016) Part 35—Medical use of byproduct material. United States Nuclear Regulatory Commission. http://www.nrc.gov/reading-rm/doc-collections/cfr/part035/
41. Delacroix D, Guerre JP, Leblanc P, Hickman C (2002) Radionuclide and radiation protection data handbook, 2nd edn. Nuclear Technology Publishing, Ashford
42. DACTARI CEA (2016) DACTARI: DAtabase for chemical toxicity and radiotoxicity assessment of radionuclides. http://www.dactari.toxcea.org/. CEA Marcoule. http://www.dactari.toxcea.org/index.php?pagendx=mnu_56
43. International Atomic Energy (2014) Radiation protection and safety of radiation sources: international basic safety standards. IAEA Safety Standards Series GSR Part 3. IAEA, Vienna

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