SIXTH EDITION

AN INTRODUCTION TO RADIATION PROTECTION

ALAN MARTIN, SAM HARBISON,
KAREN BEACH AND PETER COLE

HODDER ARNOLD

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Contents

Preface

An Introduction to Radiation Protection is a comprehensive account of radiation hazards and their control. The book is intended to meet the requirements of a wide range of readers who are involved, either directly or indirectly, with ionizing radiation, including nuclear plant designers and operators, medical practitioners and technicians, dentists, and research workers, In particular, we believe that the work is suitable for the health physics surveyors and technicians who are concerned with the day-to-day control of radiation hazards in nuclear power stations, research establishments, hospitals and in industry. In the UK, the accepted standards and curricula for formal training and qualification in radiation protection are included at Level 2 in the system of National and Scottish Vocational Qualifications (N/SVQ) and the early chapters dealing with the scientific background and the general principles of radiation protection are aimed at this level. The later chapters dealing with particular aspects of the subject are more detailed so that, for example, health physics surveyors in a nuclear power station or technicians in a hospital can get a deeper understanding of the problems in their own areas. Experience with previous editions has shown that, as a result of this structure, the book is also widely used for training at the more advanced N/SVQ Levels 3 and 4, and in many post-graduate courses.

In the Chapter on legislation and regulations relating to radiation protection, while we have concentrated mainly on the requirements in the UK, we have also included brief summaries of the requirements in a number of other countries. Otherwise the principles and practice set out in the book are applicable internationally. Although SI units are used throughout, a table of conversion factors is provided for those still using 'old' units.

Every attempt has been made to avoid detailed mathematical treatment but it has been necessary, in some areas, to use some simple mathematics. This includes squares, square roots, exponentials, logarithms and the plotting of graphs on logarithmic scales. Where a mathematical treatment is used we have tried to present it in such a way that, if the mathematics is not fully understood, it does not preclude an understanding of the chapter in general.

As far as possible each chapter is self-contained so that the reader can find all the information on a particular aspect without having to search through several chapters. The early chapters deal with basic physical principles, the nature of the hazard arising from the interaction of ionizing radiations with biological systems and the levels of radiation which are regarded as acceptable. Later chapters deal with the methods of measurement and control which are applied to attain these levels. In the second half of the book there are individual chapters on the more specialized topics of radiation protection in the nuclear industry, radioactive waste and the decommissioning of radioactive facilities, radiation protection in the non-nuclear industry, radiation protection in medicine and

radiological emergencies. There are also Chapters on legislation, risk assessment and on the organization of radiation protection services. Each chapter is followed by a summary in note form, in which the key points are reiterated. In addition, revision questions, requiring both descriptive and numerical answers, are provided for the majority of chapters.

Although we have benefited from many helpful and constructive suggestions during the preparation of this and earlier editions, the opinions and conclusions expressed in the book are those of the authors.

AM
SAH
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London, 2012

Companion website

This book has a companion website available at:

www.hodderplus.com/radiationprotection

The website contains downloadable images from the book as well as sample lectures in Powerpoint.

To access these resources please register on the website using the following access details:

Serial number: **756rp32jw9jk**

Once you have registered you will not need the serial number but can log in using the username and password that you will create during your registration.

1 The structure of matter

1.1 INTRODUCTION

Matter is the name given to the materials of which the Universe is composed. It exists in three physical forms: solid, liquid or gas. All matter consists of a number of simple substances called elements.

An **element** is a substance that cannot be broken down by ordinary chemical processes into simpler substances. There are 92 naturally occurring elements, some examples of which are carbon, oxygen, iron and lead. Another 20 or so have been produced artificially over the past 70 years or so, the best known of which is plutonium.

In nature, elements are usually chemically linked to other elements in the form of compounds. A **compound** consists of two or more elements chemically linked in definite proportions, for example water, H_2O, which consists of two atoms of hydrogen and one atom of oxygen.

1.2 THE ATOM

Consider an imaginary experiment in which a quantity of some element is subjected to repeated subdivisions. Using ordinary optical instruments, a stage would eventually be reached when the fragments would cease to be visible. Supposing, however, that suitable tools and viewing apparatus were available, would it be possible to repeat the divisions of the original element indefinitely, or would a stage be reached where the matter can no longer be subdivided?

More than 2000 years ago, Greek philosophers considered this question. With none of our modern instruments available to them, all they could do was consider the problem in a logical manner. From this philosophical approach, some of them decided that eventually a limit must be reached. They called the individual particles of matter which could not be further subdivided **atoms**. It was also postulated by some of the philosophers that all substances consist of these same atoms, different arrangements of the constituent atoms giving the different properties of the substances and the density being determined by how tightly the atoms are packed.

Early in the nineteenth century, an atomic theory with a scientific basis was advanced which confirmed many of the views held by the ancient philosophers. This was the atomic theory of Dalton, which was able to explain the well-established but little-understood chemical laws. Modern theory has diverged somewhat from Dalton's but he did establish the principle that matter consists of atoms, each element having its own characteristic atom.

1.3 THE STRUCTURE OF THE ATOM

It is now known that atoms are not solid, indivisible objects as the Greek philosophers believed but are composed of even smaller particles. These particles, from which all atoms are constructed, are called **protons**, **neutrons** and **electrons**.

The **proton** (p) carries a positive electrical charge of magnitude 1 unit on the nuclear scale, and a mass of approximately 1 atomic mass unit (u).

The **electron** (e^-) has a negative electrical charge of the same magnitude as the proton's positive charge. It has a mass of 1/1840 u, which, for most purposes, is neglected when considering the mass of the atom.

The **neutron** (n) is electrically neutral and has a mass of approximately 1 u. It can be helpful in understanding the processes of radioactive decay (see Chapter 2, Radioactivity and radiation) to regard the neutron as a close combination of a proton and an electron, with the positive charge of the proton being cancelled by the negative charge of the electron.

It should be realized that the charges and masses on the nuclear scale are extremely small. The electrical charge associated with a proton or electron is about 1.6×10^{-19} coulombs (C). For comparison, the charge stored in a typical AA-type battery is about 5000 C. Similarly, 1 u corresponds to about 1.7×10^{-27} kg.

The neutrons and protons of an atom form a central core or nucleus, around which the electrons occupy various orbits, normally referred to as shells. The shell closest to the nucleus can contain a maximum of two electrons, the second can have up to eight, and the outer shells have progressively greater numbers. The inner shell is known as the **K shell**, the second is called the **L shell**, the third the **M shell**, the fourth the **N shell** and so on. The maximum numbers of electrons in the K, L, M and N shells are 2, 8, 18 and 32, respectively. For example, the atomic system of zinc, illustrated in Figure 1.1, has 30 electrons arranged in four shells.

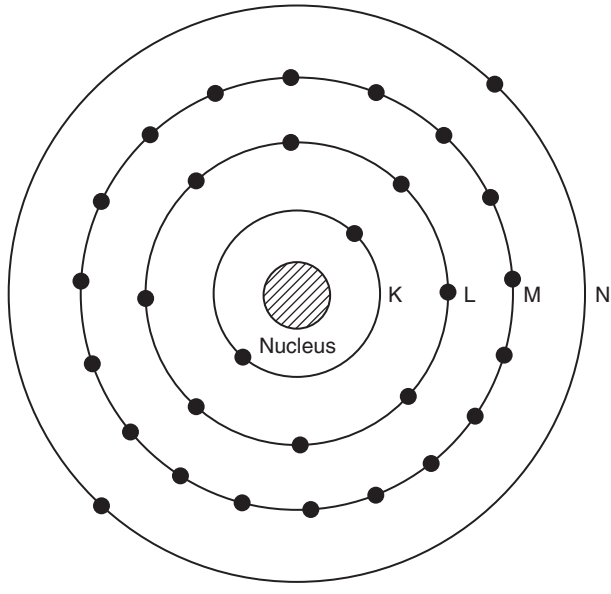


Figure 1.1 The atomic system of zinc.

Each atom normally has the same number of protons as electrons. This means that the total positive charge on the nucleus is equal to the total negative charge of the atomic electrons, and so the atom is normally electrically neutral. Two simple atoms, those of hydrogen and helium, are illustrated in Figure 1.2. Note that this particular hydrogen atom is the only atom that does not contain neutrons.

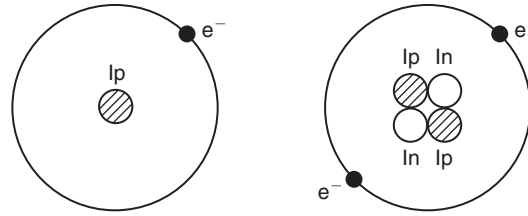


Figure 1.2 The atomic systems of hydrogen and helium.

1.4 ELEMENTS AND ATOMIC NUMBER

The number of protons in an atom is called the atomic number and is represented by the symbol Z:

$$\text{Atomic number } (Z) = \text{number of protons}$$

For example, hydrogen has one proton, $Z=1$, and helium has two protons, $Z=2$. Note that it is the electrons in an atom that determine its chemical properties. In an electrically neutral atom, the number of protons equals the number of electrons, and so, indirectly, it is the number of protons in an atom that defines the element.

Thus:

all atoms with an atomic number of 1 are hydrogen atoms (chemical symbol H)

all atoms with an atomic number of 3 are lithium atoms (Li)

all atoms with an atomic number of 4 are beryllium atoms (Be)

all atoms with an atomic number of 5 are boron atoms (B)

all atoms with an atomic number of 6 are carbon atoms (C) …, etc.,

up to the heaviest naturally occurring element, uranium (U), which has an atomic number of 92. About 20 or so elements of a higher atomic number have been artificially produced in the past 70 years or so. They are all unstable and can only be made under special conditions which are not found naturally on Earth.

1.5 ISOTOPES AND MASS NUMBER

Although all the atoms of a particular element contain the same number of protons, they may occur with different numbers of neutrons. This means that an element can have several types of atom. For example, hydrogen can occur with zero, one or two neutrons in its nucleus, and the three different types of atom are called **isotopes** of hydrogen.

The mass of an atom is determined by the number of protons and neutrons if the very small mass of the atomic electrons is neglected. The sum of the number of protons plus the number of neutrons is called the **mass number** and is represented by the symbol *A*:

$$\text{Mass number } (A) = \text{number of protons} + \text{number of neutrons}$$

For example, the helium atom in Figure 1.2b contains two protons and two neutrons and so has a mass number of 4. Helium can also occur with one or three neutrons in the nucleus, as shown in Figure 1.3. These three isotopes are normally referred to as helium-3, helium-4 and helium-5, which are often shortened to He-3, He-4 and He-5.

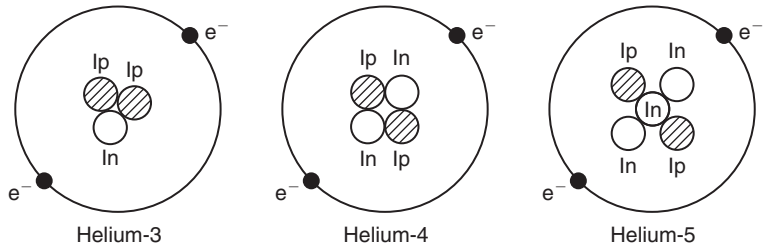


Figure 1.3 The three isotopes of helium.

In symbolic form, an isotope can also be written as ${}^{A}_{Z}\text{X}$, where X is the symbol for the element. In this format, helium-3 is written ${}^{3}_{2}\text{He}$. Strictly, showing the atomic number is unnecessary because the name of the element defines the atomic number, so in most cases it is sufficient to write this as ${}^{3}\text{He}$. Throughout this text, in using this notation, the atomic number is included where it assists an understanding of the topic.

Considering another example, the element phosphorus (P) has an atomic number of 15 (i.e. each atom contains 15 protons), but it can occur with different numbers of neutrons. In other words, there are several isotopes of phosphorus, as shown below.

${}^{28}_{15}\text{P}$ has 15 protons and 13 neutrons ($Z=15, A=28$)

${}^{29}_{15}\text{P}$ has 15 protons and 14 neutrons ($Z=15, A=29$)

${}^{30}_{15}\text{P}$ has 15 protons and 15 neutrons ($Z=15, A=30$)

${}^{31}_{15}\text{P}$ has 15 protons and 16 neutrons ($Z=15, A=31$)

${}^{32}_{15}\text{P}$ has 15 protons and 17 neutrons ($Z=15, A=32$)

${}^{33}_{15}\text{P}$ has 15 protons and 18 neutrons ($Z=15, A=33$)

${}^{34}_{15}\text{P}$ has 15 protons and 19 neutrons ($Z=15, A=34$)

It is important to note that all the isotopes of a given element are **chemically** identical since the chemical properties are determined by the atomic number of the element.

Most elements occur naturally as a mixture of isotopes, and other isotopes may be produced by bombarding a naturally occurring isotope with nuclear particles, for example by neutrons in a nuclear reactor. These artificially produced isotopes are unstable and will eventually disintegrate with the emission of a secondary particle (see Chapter 2).

Apart from the few lightest elements, the number of neutrons exceeds the number of protons in an atom. The difference becomes greater as the atomic number increases, as illustrated by the following examples:

$^{4}_{2}$He has two protons and two neutrons

$^{31}_{15}$P has 15 protons and 16 neutrons

$^{65}_{30}$Zn has 30 protons and 35 neutrons

$^{238}_{92}$U has 92 protons and 146 neutrons

Data on the known isotopes of all the elements, both naturally occurring and artificially produced, have been arranged systematically in a table known as the **chart of the nuclides**, which will be discussed in more detail in Chapter 2. The term **nuclide** means any isotope of any element.

1.6 ANCIENT AND MODERN THEORIES

It can now be seen that the ancient Greek philosophers were remarkably close to the truth in their theory that all substances are constituted from the same basic particles. However, instead of being different arrangements of only one type of particle, different substances appear to result from various combinations of protons, neutrons and electrons. It is now known that protons and neutrons are made up of even smaller particles called quarks, and there is some evidence of apparently more fundamental particles. Thus, the ancient Greeks may yet prove to have been right in their conjecture that there is just one fundamental particle that provides the basis for all others.

SUMMARY OF KEY POINTS

Element: material whose atoms all have the same number of protons.
u: atomic mass unit.
Proton: atomic particle, mass 1 u, charge +1 unit.
Electron: atomic particle, mass 1/1840 u, charge −1 unit.
Neutron: close combination of proton and electron, mass 1 u, electrically neutral.
Atom: central nucleus of protons and neutrons, around which electrons occupy orbits.
Atomic number (*Z*): number of protons.
Mass number (*A*): number of protons plus number of neutrons.
Isotope: one of several nuclides with the same atomic number.
Notation: there are several ways of referring to an isotope: for example, phosphorus-32, P-32, $^{32}_{15}$P and ^{32}P.
Nuclide: a nuclear species.

REVISION QUESTIONS

1. Following the illustration in Figure 1.1, draw an atom of each of the following nuclides: a) $^{4}_{2}$He, b) $^{6}_{3}$Li, c) $^{7}_{3}$Li, d) $^{8}_{4}$Be
2. For each of the following nuclides of helium, sodium and uranium, how many protons, electrons and neutrons are there in their atoms: $^{4}_{2}$He, $^{23}_{11}$Na, $^{238}_{92}$U?
3. What are the masses and charges on the atomic scale of protons, electrons and neutrons?
4. Which atomic property determines the chemical behaviour of an element?
5. Explain what is meant by the term isotope. Give some examples.

2 Radioactivity and radiation

2.1 INTRODUCTION

It is found that a few naturally occurring substances consist of atoms which are unstable – that is, they undergo spontaneous transformation into more stable product atoms. Such substances are said to be **radioactive**, and the transformation process is known as **radioactive decay**. Radioactive decay is usually accompanied by the emission of radiation in the form of charged particles and gamma (γ) rays.

The fact that some elements are naturally radioactive was first realized by Becquerel in 1896. He observed the blackening of photographic emulsions in the vicinity of a uranium compound. This was subsequently attributed to the effect of radiation being emitted by the uranium. In the following 10 years, the experimental work of Rutherford and Soddy, Pierre and Marie Curie and others established the fact that some types of nuclei are not completely stable. These unstable nuclei decay and emit radiations of three main types, called alpha, beta and gamma radiation.

2.2 ALPHA, BETA AND GAMMA RADIATION

Alpha (α) radiation was shown by Rutherford and Royds to consist of helium nuclei, which themselves consist of two protons and two neutrons. These four particles are bound together so tightly that the α particle behaves in many situations as if it were a fundamental particle. An α particle has a mass of 4 units and carries 2 units of positive charge.

Beta (β) radiation consists of high-speed electrons which originate in the nucleus. These 'nuclear electrons' have identical properties to the atomic electrons, that is they have a mass of 1/1840 u and carry one unit of negative charge. Another type of β radiation was discovered by C. D. Anderson in 1932. This consists of particles of the same mass as the electron but with one unit of positive charge; it is known as **positron radiation**. Although less important from a radiation protection viewpoint than negative β particles, a knowledge of positrons is necessary in order to understand certain radioactive decay mechanisms. Beta radiation is signified β^- (electrons) or β^+ (positrons). In everyday use, the term β radiation normally refers to the negative type, β^-.

Gamma (γ) radiation, although somewhat analogous to α or β particles, is electromagnetic in nature and can be described as consisting of 'particles' called '**photons**'. However, photons do not have any mass or electrical charge, and instead they consist of packets (or 'quanta') of energy transmitted in the form of a wave motion (wave packets). Other well-known members of this class of radiation are radio waves and visible light. The

amount of energy in each quantum is related to the wavelength of the radiation. The energy is inversely proportional to the wavelength, which means that the shorter the wavelength the higher the energy. Mathematically, this is written as $E \propto 1/\lambda$, where E is the energy of the quantum or photon of electromagnetic radiation and λ is its wavelength. Another class of electromagnetic radiation which is in most respects identical to γ radiation is known as X radiation. The essential difference between the two types of radiation lies in their origin. Whereas γ-rays result from changes in the nucleus, X-rays are emitted when atomic electrons undergo a change in orbit.

The wavelength of electromagnetic radiation varies over a very wide range, as illustrated in Table 2.1.

All electromagnetic radiations travel through free space with the same velocity of 3×10^8 m/s. Their velocity decreases in dense media, but in air the decrease is negligible.

Table 2.1 Wavelengths of electromagnetic radiations

Type of radiation	Wavelength, λ (m)
Radio waves (long wave)	1500
Radio waves (very high frequency)	3
Visible light	10^{-6}–10^{-7}
X-rays, 50 keV energy	2.5×10^{-11}
γ-rays, 1 MeV energy	1.2×10^{-12}

2.3 THE ELECTRONVOLT

Radiation energy is expressed in **electronvolts** (eV). One electronvolt is the energy gained by an electron in passing through an electrical potential of 1 volt. For example, in an X-ray tube, electrons are accelerated from a heated tungsten filament through an electrical potential of typically 100 000 volts to the anode. The electrons therefore have an energy of 100 000 eV when they strike the anode.

The electronvolt is a very small unit, so radiation energies are usually expressed in **kilo** (1000) or **mega** (1 000 000) electronvolts:

One kiloelectronvolt = 1 keV = 1000 eV

One megaelectronvolt = 1 MeV = 1000 keV = 1 000 000 eV

The radiation energies of interest in radiation protection are generally in the range of 100 keV to 10 MeV. It is important to appreciate that even if the radiation being considered is not β (electron) radiation, it is still possible to express its energy in terms of the electronvolt.

The energy of a particle depends on its mass and velocity; for example, the kinetic energy (E_K) of a particle of mass (m) travelling with velocity (v) which is much smaller than the velocity of light is given by the equation:

$$E_K = \tfrac{1}{2}mv^2$$

(A correction is necessary for particles which have velocities approaching the velocity of light.) A small particle such as an electron requires a much higher velocity than, say, an α particle in order to have the same kinetic energy.

In the case of electromagnetic radiation, the energy is inversely proportional to the wavelength of the radiation. Thus, radiations with short wavelengths have higher energies than radiations with longer wavelengths. Note that all electromagnetic radiation travels at the speed of light.

2.4 THE MECHANISM OF RADIOACTIVE DECAY

The nuclei of the heavier elements found in nature are so large that they are slightly unstable. For example, the isotope uranium-238 has 92 protons and 146 neutrons. To achieve greater stability, the nucleus may emit an α particle, thus reducing its numbers of protons and neutrons to 90 and 144, respectively. This means that the nucleus now has an atomic number (Z) of 90 instead of 92 and so is no longer a uranium nucleus. It is now an isotope of the element thorium (Th), with an atomic number of 90 and a mass number of 234, namely thorium-234 (^{234}Th). This decay process may be represented as follows:

$$^{238}_{92}\mathrm{U} \longrightarrow {}^{4}_{2}\alpha + {}^{234}_{90}\mathrm{Th}$$

or, more commonly:

$$^{238}_{92}\mathrm{U} \xrightarrow{\alpha} {}^{234}_{90}\mathrm{Th}$$

Another example of this process is the decay of polonium-218 (^{218}Po) by α emission to lead-214 (^{214}Pb):

$$^{218}_{84}\mathrm{Po} \xrightarrow{\alpha} {}^{214}_{82}\mathrm{Pb}$$

It was pointed out in Chapter 1 that there are more neutrons than protons in heavy nuclei. α emission reduces the number of each by two, but the proportionate reduction is considerably less for neutrons than for protons. In the ^{238}U decay process, the number of protons is reduced by 2 out of 92, whereas the number of neutrons is reduced by 2 out of 146, which is, proportionately, significantly less. The effect of α emission is therefore to produce neutron-rich nuclei that are still unstable. The nucleus does not simply eject a neutron (or neutrons) to correct this instability. Instead, one of the neutrons in the nucleus changes into a proton by emitting a β particle, that is, a high-speed electron:

$$^{1}_{0}\mathrm{n} \longrightarrow {}^{1}_{1}\mathrm{p} + \beta^{-}$$

This phenomenon is known as **β emission**. In the case of ^{234}Th, formed by the α decay of ^{238}U, the nucleus further decays by β emission to protactinium-234 (^{234}Pa):

$$^{234}_{90}\mathrm{Th} \longrightarrow {}^{234}_{91}\mathrm{Pa} + \beta^{-}$$

or:

$$^{234}_{90}\mathrm{Th} \xrightarrow{\beta^{-}} {}^{234}_{91}\mathrm{Pa}$$

Considering again polonium-218, the complete decay is:

$$^{218}_{84}\text{Po} \xrightarrow{\alpha} {}^{214}_{82}\text{Pb} \xrightarrow{\beta^-} {}^{214}_{83}\text{Bi}$$

The resulting atom is bismuth-214, which is also unstable and so further α- and β-decay processes occur until a stable atom is produced.

Electrons emitted during β decay have a continuous distribution in energy, ranging from zero to a maximum energy (E_{max}), which is characteristic of the particular nucleus. The most probable β energy is about $\frac{1}{3}E_{max}$ (see Fig. 2.1).

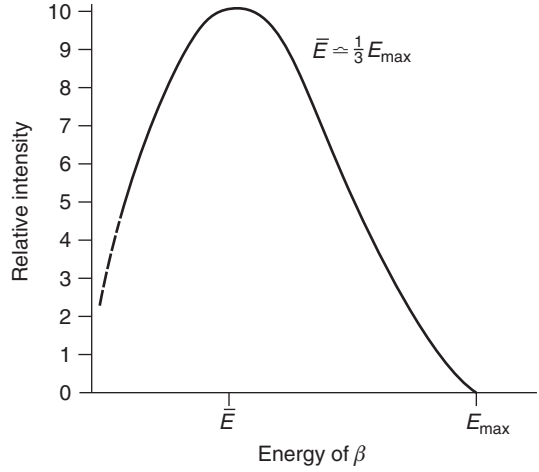


Figure 2.1 Typical β spectrum.

In most cases, after the emission of an α or β particle, the nucleus rearranges itself slightly, releasing energy by **γ emission**.

Two other decay processes should also be mentioned, namely **positron emission** and **electron capture**. In positron emission, a proton in the nucleus ejects a positive electron (β^+) and so becomes a neutron:

$$^{1}_{1}\text{p} \longrightarrow {}^{1}_{0}\text{n} + \beta^+$$

For example, sodium-22 (^{22}Na) decays by positron emission to neon-22:

$$^{22}_{11}\text{Na} \xrightarrow{\beta^+} {}^{22}_{10}\text{Ne}$$

Electron capture is a process in which an electron from an inner orbit is captured by the nucleus, resulting in the conversion of a proton to a neutron:

$$^{1}_{1}\text{p} + \text{e}^- \longrightarrow {}^{1}_{0}\text{n}$$

A rearrangement of atomic electrons then causes the emission of X-rays.

2.5 NATURAL RADIOACTIVE SERIES

Apart from ^{22}Na, the above examples of radioactive decay are all naturally occurring radioactive substances and belong to the so-called **natural radioactive series**. There are three natural radioactive series, called the thorium, uranium–radium and actinium series (see Table 2.2). Also included in this table is the neptunium series, the longest member of which has a half-life ($T_{½}$) of 2.2×10^6 years. This is much less than the age of the Earth and so the series has long since decayed. However, neptunium-237 is produced artificially in nuclear reactors and can be important in some situations.

Table 2.2 Natural radioactive series

Series name	Final stable nucleus	Longest-lived member
Thorium	^{208}Pb	^{232}Th ($T_{1/2} = 1.39 \times 10^{10}$ years)
Uranium–radium	^{206}Pb	^{238}U ($T_{1/2} = 4.50 \times 10^{9}$ years)
Actinium	^{207}Pb	^{235}U ($T_{1/2} = 8.52 \times 10^{8}$ years)
Neptunium	^{209}Bi	^{237}Np ($T_{1/2} = 2.20 \times 10^{6}$ years)

The term 'series' is used because an atom undergoes a succession of radioactive transformations until it reaches a stable state. In the thorium series, the atom is initially thorium-232 and undergoes a series of radioactive decays as follows:

$$^{232}_{90}\text{Th} \xrightarrow{\alpha} {}^{228}_{88}\text{Ra} \xrightarrow{\beta^-} {}^{228}_{89}\text{Ac} \xrightarrow{\beta^-} {}^{228}_{90}\text{Th} \xrightarrow{\alpha} {}^{224}_{88}\text{Ra} \xrightarrow{\alpha}$$

$$^{220}_{86}\text{Rn} \xrightarrow{\alpha} {}^{216}_{84}\text{Po} \xrightarrow{\alpha} {}^{212}_{82}\text{Pb} \xrightarrow{\beta^-} {}^{212}_{83}\text{Bi} \xrightarrow{\beta^-} {}^{212}_{84}\text{Po} \xrightarrow{\alpha} {}^{208}_{82}\text{Pb}$$

The half-lives of these members of the decay series range from 0.15 s for polonium-216 to about 1.4×10^{10} years for thorium-232.

2.6 INDUCED RADIOACTIVITY

Lighter elements may be made radioactive by bombarding them with nuclear particles. One such process involves the bombardment of stable nuclei of an element by neutrons in a nuclear reactor. A neutron may be captured by a nucleus with the emission of a γ photon. This is known as a neutron, gamma (n, γ) reaction. The resulting atom is usually unstable because of the excess neutron and will eventually decay by β emission.

Thus, if the stable isotope cobalt-59 is bombarded or irradiated with neutrons, atoms of the radioactive isotope cobalt-60 are produced. These atoms will eventually undergo β decay and become atoms of the stable isotope nickel-60. This process is written as:

$$^{59}_{27}\text{Co}(\text{n},\gamma)^{60}_{27}\text{Co} \xrightarrow{\beta^-} {}^{60}_{28}\text{Ni}$$

There are various other activation and decay processes, which will be discussed later.

2.7 THE UNIT OF RADIOACTIVITY

The decay of a radioactive sample is statistical in nature and it is impossible to predict when any particular atom will disintegrate. The consequence of this random behaviour of radioactive atoms is that the radioactive decay law is exponential in nature, and is expressed mathematically as:

$$N_t = N_0 e^{-\lambda t}$$

where N_0 is the number of nuclei present initially, N_t is the number of nuclei present at time t and λ is the radioactive decay constant.

The **half-life** ($T_{½}$) of a radioactive species is the time required for one-half of the nuclei in a sample to decay. It is obtained by putting $N_t = N_0/2$ in the above equation:

$$N_0/2 = N_0 e^{-\lambda T_{½}}$$

dividing across by N_0 and taking logs:

$$\log_e(½) = -\lambda T_{½}$$

Now:

$$\log_e(½) = -\log_e(2)$$

and so:

$$T_{½} = \frac{\log_e 2}{\lambda} = \frac{0.693}{\lambda}$$

Since the disintegration rate, or **activity**, of the sample is proportional to the number of unstable nuclei, this also varies exponentially with time in accordance with the equation:

$$A_t = A_0 e^{-\lambda t}$$

This relationship is illustrated in Figure 2.2, which shows the variation of sample activity with time. In one half-life the activity decays to $½A_0$, in two half-lives to $¼A_0$, and so on. In practice, the decay of a sample is usually plotted on a log-linear graph in which the vertical axis has a logarithmic scale. The plot then becomes a straight line with a slope that depends on the half-life of the sample. This is illustrated in more detail in section 10.2.3. The half-life of a particular radioactive isotope is constant and its measurement assists in the identification of radioactive samples of unknown composition. This method can be applied only to isotopes whose disintegration rates change appreciably over reasonable counting periods. At the other end of the scale, the isotope must have a long enough half-life to allow some measurements to be made before it all disintegrates. To determine extremely short and extremely long half-lives, more elaborate means must be used. Half-lives range from about 10^{-14} years (^{212}Po) to about 10^{17} years (^{209}Bi), which represents a factor of 10^{31}.

For many years, the unit of radioactivity was the **curie** (Ci), but this has now been generally replaced by the SI (Système International d'Unités) unit, the **becquerel** (Bq).

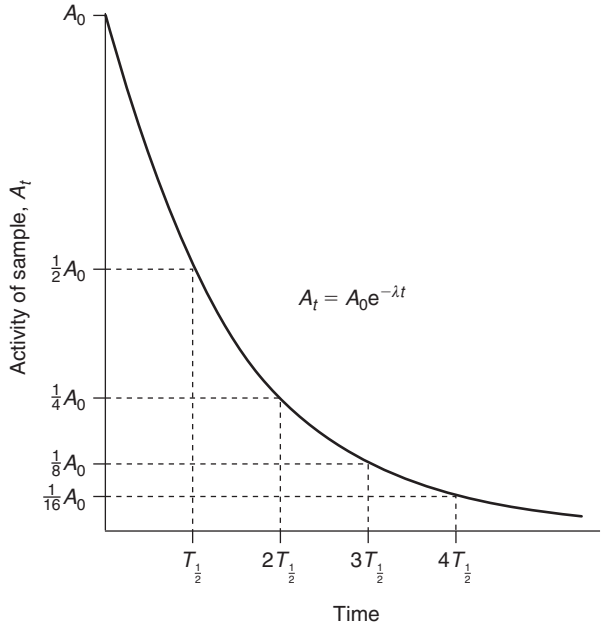


Figure 2.2 Variation of activity with time.

The curie was originally related to the activity of 1 g of radium, but the definition was later standardized as 3.7×10^{10} nuclear disintegrations (dis)/s, which is almost the same:

$$1\,\text{Ci} = 3.7 \times 10^{10}\,\text{dis/s or } 2.22 \times 10^{12}\,\text{dis/min}$$

The becquerel is defined as one nuclear disintegration per second and, compared with the curie, it is a very small unit. In practice, most radioactive sources are much larger than the becquerel and the following multiplying prefixes are used to describe them:

1 becquerel (Bq) = 1 dis/s

1 kilobecquerel (kBq) = 10^3 Bq = 10^3 dis/s

1 megabecquerel (MBq) = 10^6 Bq = 10^6 dis/s

1 gigabecquerel (GBq) = 10^9 Bq = 10^9 dis/s

1 terabecquerel (TBq) = 10^{12} Bq = 10^{12} dis/s

1 petabecquerel (PBq) = 10^{15} Bq = 10^{15} dis/s

For simplicity, in this text only Bq, MBq and TBq have been used.

As explained earlier, a disintegration usually involves the emission of a charged particle (α or β). This may be accompanied, although not always, by one or more γ emissions. Some nuclides emit only X or γ radiation.

2.8 THE NUCLIDE CHART

The **nuclide chart** is a compilation of information on all known stable and unstable nuclides and a portion of it is reproduced in Figure 2.3.

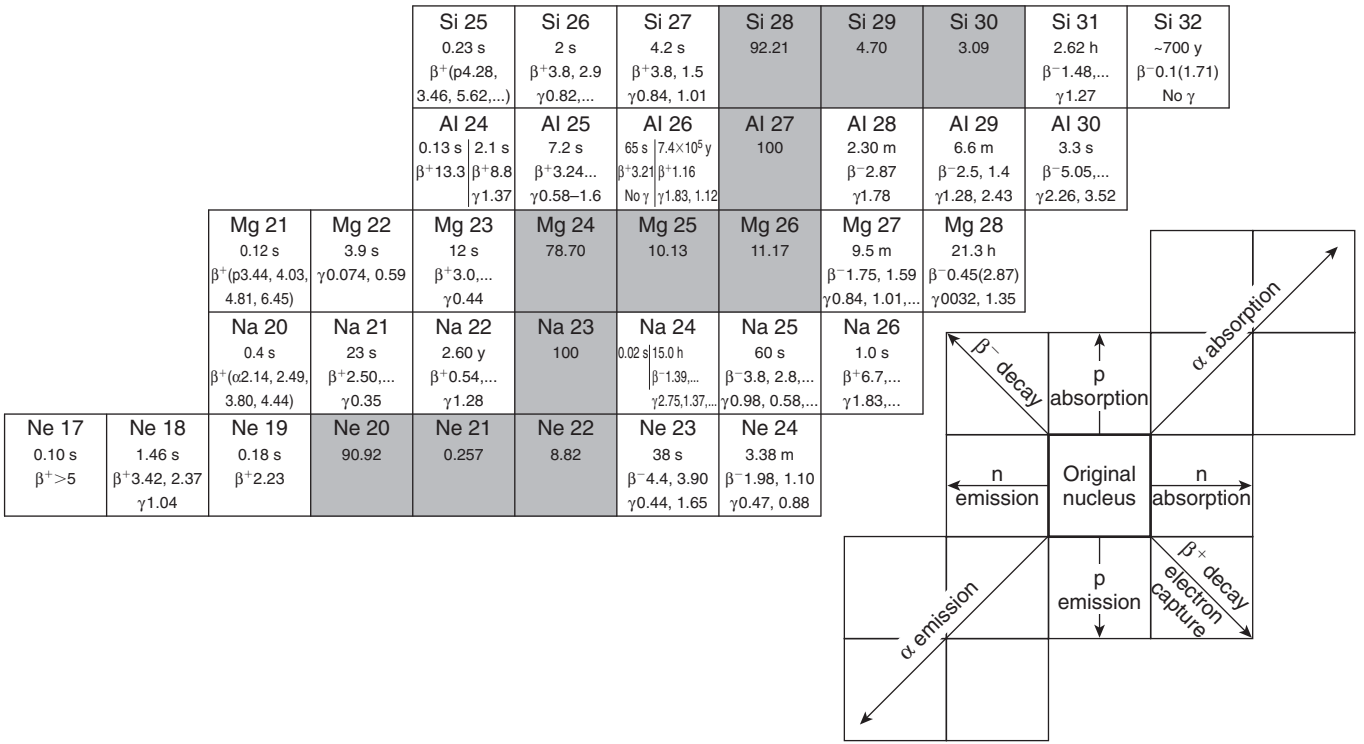


Figure 2.3 A portion of the nuclide chart.

In the chart, each horizontal line represents an element and the squares on that line represent the nuclides or isotopes of the element. Relevant information regarding the nuclide is printed inside the square. Stable, naturally radioactive and artificial nuclides are differentiated by the use of different colours or shading of the squares. In each case the symbol and mass number are shown as well as the natural abundance of the isotope. For radioactive isotopes, the half-life, the mode or modes of decay, and the main energies of the emitted particles or γ rays are shown. In the chart illustrated in Figure 2.3, all the nuclides on the same horizontal line have the same atomic number, while all nuclides with the same mass number lie on a 45° diagonal line running from the upper left to the lower right. Many nuclide charts contain additional information which has been omitted from the sample chart shown in Figure 2.3 for the sake of clarity.

Also shown at the lower right of the figure is the effect on the original nucleus of various capture or decay reactions. For example, an n, γ reaction on a nucleus moves it one space to the right on the same row. Thus, an n, γ reaction on sodium-23 (^{23}Na) produces sodium-24 (^{24}Na). This ^{24}Na decays with a half-life of 15.0 h by emitting β^- particles of 1.39 MeV and γ rays of 2.75 MeV and 1.37 MeV. The nucleus resulting from the decay of ^{24}Na is magnesium-24 (^{24}Mg), which is stable.

It can be seen that the nuclide chart is an extremely valuable source of information on the properties of both stable and unstable nuclides. A complete chart with detailed information can be found at http://www-nds.iaea.org/relnsd/vchart/index.html.

2.9 INTERACTION OF RADIATION WITH MATTER

2.9.1 Charged particles

Both α and β particles lose energy mainly through interactions with atomic electrons in the absorbing medium. The energy transferred to the electrons causes them either to be excited to a higher energy level (excitation) or separated entirely from the parent atom (ionization). Another important effect is that when charged particles are slowed down very rapidly, they emit energy in the form of X-rays. This is known as **bremsstrahlung** (braking radiation) and is of practical importance only in the case of β radiation.

2.9.2 X and γ radiations

X and γ radiations interact with matter through a variety of alternative mechanisms, the three most important of which are the **photoelectric effect**, **Compton scattering** and **pair production**. In the photoelectric effect, all the energy of a X- **or** γ photon is transferred to an atomic electron which is ejected from its parent atom. The photon is, in this case, completely absorbed. Conversely, when Compton scattering occurs, only part of the energy of the photon is transferred to an atomic electron. The scattered photon then continues with reduced energy.

In the intense electric field close to a charged particle, usually a nucleus, an energetic γ photon may be converted into a positron–electron pair. This is pair production, and the two resulting particles share the available energy.

Thus, all three interactions result in the photon energy being transferred to atomic electrons which subsequently lose energy, as described in section 2.9.1.

2.9.3 Neutrons

Neutrons are uncharged and cannot cause ionization directly. Neutrons ultimately transfer their energy to charged particles. Also a neutron may be captured by a nucleus, usually resulting in γ emission. These processes are described in greater detail in section 8.5. Table 2.3 summarizes the types of interactions of nuclear radiations with matter.

Table 2.3 Interactions of nuclear radiations

Radiation	Process	Remarks
α	Collisions with atomic electrons	Leads to excitation and ionization
β	(a) Collisions with atomic electrons	Leads to excitation and ionization
	(b) Slowing-down in field of nucleus	Leads to emission of bremsstrahlung
X and γ radiation	(a) Photoelectric effect	Photon is completely absorbed
	(b) Compton effect	Only part of the photon energy is absorbed
	(c) Pair production	
Neutron	(a) Elastic scattering (b) Inelastic scattering (c) Capture processes	Discussed in Chapter 8

2.10 PENETRATING POWERS OF NUCLEAR RADIATIONS

The α particle is a massive particle (by nuclear standards) that travels relatively slowly through matter. It thus has a high probability of interacting with atoms along its path and will give up some of its energy during each of these interactions. As a consequence, α particles lose their energy very rapidly and travel only very short distances in dense media.

Beta particles are very much smaller than α particles and travel much faster. They thus undergo fewer interactions per unit length of track and give up their energy more slowly than α particles. This means that β particles travel further in dense media than α particles.

Gamma radiation loses its energy mainly by interacting with atomic electrons. It travels very large distances even in dense media and is very difficult to absorb completely.

Neutrons give up their energy through a variety of interactions, the relative importances of which are dependent on the neutron energy. For this reason, it is common practice to divide neutrons into at least three energy groups: fast, intermediate and thermal. Neutrons are very penetrating and will travel large distances even in dense media.

The properties and ranges of the various nuclear radiations are summarized in Table 2.4. The ranges are only approximate since they depend on the energy of the radiation.

Table 2.4 Properties of nuclear radiations

Radiation	Mass (u)	Charge	Range in air	Range in tissue
α	4	+2	~0.03 m	~0.04 mm
β	1/1840	-1 (positron +1)	~3 m	~5 mm
X and γ radiation	0	0	Very large	Through body
Fast neutron	1	0	Very large	Through body
Thermal neutron	1	0	Very large	~0.15 m

SUMMARY OF KEY POINTS

Radioactive decay: transformation of an unstable atomic nucleus into a more stable one, usually accompanied by the emission of charged particles and γ-rays.

Alpha (α) radiation: helium nuclei, two protons and two neutrons, mass 4 units, charge +2 units.

Beta (β) radiation: high-speed electrons which originate in the nucleus, mass 1/1840 u, charge −1 (electron) or +1 (positron).

Gamma (γ) radiation: electromagnetic radiation, very short wavelength, $E \propto 1/\lambda$, mass 0, charge 0.

Electronvolt: energy gained by an electron in passing through an electric potential of 1 volt.

$$10^6\,\text{eV} \equiv 10^3\,\text{keV} \equiv 1\,\text{MeV}$$

Natural radioactive series consist of naturally occurring radioactive substances; the three series are thorium, uranium–radium and actinium.

Induced radioactivity: radioactivity caused by bombarding stable atoms with nuclear particles, for example by neutrons in a nuclear reactor.

Radioactive decay law:

$$N_t = N_0 e^{-\lambda t}$$

Half-life: time required for one half of the nuclei of a radioactive species to decay:

$$T_{1/2} = \frac{0.693}{\gamma}$$

Curie (Ci): former unit of radioactivity defined as 3.7×10^{10} dis/s.

$1\,\text{Ci} \equiv 10^3\,\text{mCi} \equiv 10^6\,\mu\text{Ci}$

Becquerel (Bq): SI unit of radioactivity, defined as 1 dis/s.

$1\,\text{TBq} \equiv 10^6\,\text{MBq} \equiv 10^{12}\,\text{Bq}$

Nuclide chart: compilation of data on all known nuclides.

Alpha particles lose energy in matter through excitation and ionization.

Beta particles lose energy by:

1. excitation and ionization of atomic electrons, and
2. rapid slowing down with emission of bremsstrahlung.

Gamma photons lose energy through:

1. photoelectric effect,
2. Compton effect, and
3. pair-production.

Neutrons lose energy through:

1. elastic scatter,
2. inelastic scatter, and
3. capture reactions.

REVISION QUESTIONS

1. Explain the difference between radioactivity and radiation.
2. Using a nuclide chart, name the products of the following radioactive decay processes:
   (a) α decay of uranium-238, $^{238}_{92}U$,
   (b) β^- decay of tritium, $^{3}_{1}H$,
   (c) β^+ decay of copper-62, $^{62}_{29}Cu$.
3. Explain why the radionuclides of the neptunium decay series are not found in nature.
4. Estimate the half-life of a radioactive sample by plotting a graph of the following series of measurements:

Time (min)	0	1	2	3	4	5	6	7	8
Activity (counts/min)	–	820	605	447	330	243	180	133	98

5. Express the following activities in megabecquerels:
   (a) 5×10^6 dis/s,
   (b) 750 kBq,
   (c) 1.3 GBq,
   (d) 6×10^7 dis/min.
6. Why is an α decay usually followed by a β^--decay?
7. Using a nuclide chart, write down the product or sequence of products which would result from an (n, γ) capture in the following nuclei:
   (a) $^{58}_{27}Fe$,
   (b) $^{23}_{11}Na$,
   (c) $^{239}_{94}Pu$.

3 Radiation units

3.1 ABSORPTION OF ENERGY

Just as heat and light transfer energy from the Sun to the Earth and the atmosphere, nuclear radiation transfers energy from a source to an absorbing medium. The source of nuclear radiation may be radioactive atoms or equipment such as X-ray machines. The effect of absorbing the more familiar types of radiation, such as heat, is to raise the temperature of the absorbing medium. If this medium is the human body, or part of it, the rise in temperature is sensed and, if it becomes excessive, avoiding action can be taken by sheltering under a sunshade (shielding), for example, or by moving further away from a fire (distance). However, a dose of gamma (γ) radiation or other nuclear radiation that is large enough to be lethal to a human being would increase the body temperature by less than one-thousandth of 1°C. The body is therefore unable to sense even very high intensities of these types of radiation.

Nuclear radiation differs from heat and other types of radiation in that each individual particle or photon has a sufficiently high energy to cause ionization. The high energy is due to the very high velocity of the particles or the short wavelength of the X and γ radiation.

3.2 IONIZATION

Ionization is the removal of an orbital electron from an atom. Since the electron has a negative charge, the atom is consequently left positively charged. The atom and the electron, so separated, are known as an **ion pair**, that is, a positive ion (the atom minus one electron) and a negative ion (the electron). To cause ionization requires energy and this is supplied by the absorption of radiation energy in the medium, which subsequently results in the production of ion pairs. The particles or photons of radiation lose their energy to the medium in the process. Figure 3.1 shows the ionization of a helium atom by an alpha (α) particle.

Normally, positive and negative ions recombine to form neutral atoms and the energy originally given to the ion pair is converted into heat energy. If the absorbing medium is a gas, such as air, the ions can be prevented from recombining by applying an electrical field. This is done by applying a voltage between two plates (electrodes) with a gas gap between them. Figure 3.2 shows such a system, known as an ionization chamber, in which ion pairs are being produced along the track of an α particle. If the applied voltage is sufficiently high, negative ions produced in the volume between the electrodes are attracted to the positive electrode and positive ions to the negative electrode. The flow of ions to the

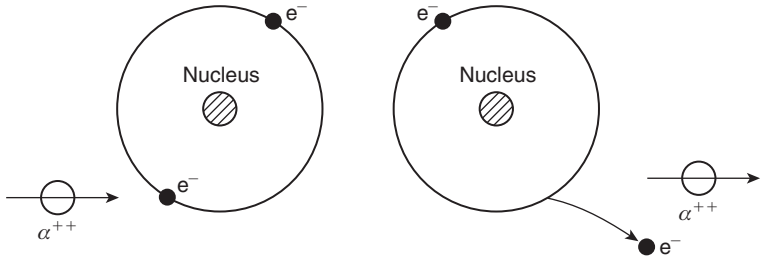


Figure 3.1 Ionization of a helium atom by an α particle.

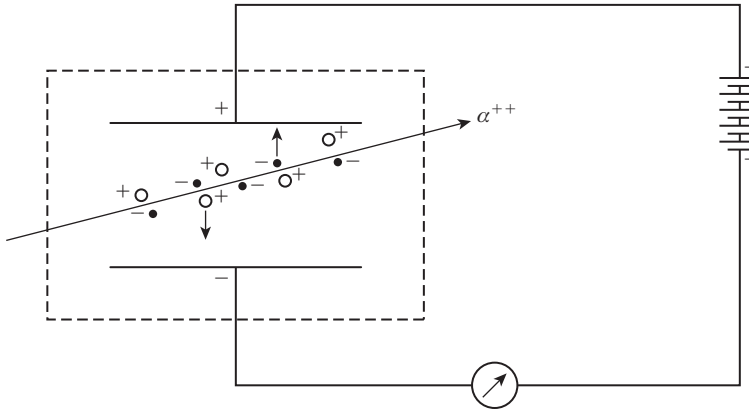


Figure 3.2 Ionization chamber system.

respective electrodes constitutes an electrical current and, since this is proportional to the intensity of radiation, ion chambers provide a means of measuring radiation. It should be realized that, although only a few ion pairs are shown in the figure, in the case of beta (β) particles, several hundreds of ion pairs are formed per centimetre of track in air and, in the case of α particles, some tens of thousands.

In a medium such as water (of which the human body is largely composed), ionization can lead to a breakdown of water molecules and the formation of chemical forms that are damaging to biological material. The harmful effects of radiation on the human system, which are described in section 4.4, are largely attributable to such chemical reactions.

As already mentioned, the ionization of a gas provides a means of detecting radiation and the first widely used radiation unit, the **roentgen**, was based on the ionizing effect on air of X and γ radiation. This unit had several limitations and so two further units, the **rad** and the **rem**, were introduced. Later, these two units were replaced in the SI system (Système International d'Unités) by the **gray** (Gy) and the **sievert** (Sv), respectively.

The gray and the sievert have been approved by the International Commission on Radiation Units and Measurements (ICRU) and used by the International Commission on Radiological Protection (ICRP). However, the older units, the rad and the rem, are still used in some countries and an explanation of the relationships between the old units and the SI units is given in Appendix A.

3.3 ABSORBED DOSE

Absorbed dose is a measure of energy deposition in any medium by any type of ionizing radiation. The original unit of absorbed dose was the rad, which was defined as an energy deposition of 0.01 joule per kilogram (J/kg).

The unit of absorbed dose in SI units is the gray and is defined as an energy deposition of 1 J/kg. Thus:

$$1\ \text{Gy} = 1\ \text{J/kg}$$

When quoting an absorbed dose, it is important to specify the absorbing medium.

3.4 EQUIVALENT DOSE

Although the quantity absorbed dose is a very useful physical concept, it transpires that, in biological systems the same degree of damage is not necessarily produced by the same absorbed dose of different types of radiation. It is found, for example, that 0.05 Gy of fast neutrons can do as much biological damage as 1 Gy of γ radiation. This difference in the radiobiological effectiveness must be taken into account if we wish to add doses of different radiations to obtain the total biologically effective dose. To do this, we must multiply the absorbed dose of each type of radiation by a **radiation weighting factor** (w_R), which reflects the ability of the particular type of radiation to cause damage. The quantity obtained when the absorbed dose is multiplied by the radiation weighting factor is known as the **equivalent dose**, H.

The unit of equivalent dose in SI units is the sievert, which is related to the gray as follows:

$$\text{equivalent dose, H (Sv)} = \text{absorbed dose (Gy)} \times w_R$$

The value of the radiation weighting factor is found to depend on the density of ionization caused by the radiation. An α particle produces about 10^6 ion pairs per millimetre of track in tissue whereas a β particle produces about 10 000/mm. The radiation weighting factor is assigned a value of 1 for γ radiation and the values for other types of radiation are related to this in accordance with their ionization densities. β radiation causes ionization of a similar density to γ radiation and so its weighting factor is also 1. The value of the radiation weighting factor for neutrons depends on the neutron energy and varies from 2.5

Table 3.1 Summary of values of radiation weighting factor

Type of radiation	Radiation weighting factor
X-rays, γ-rays and electrons	1
Protons	5
Thermal neutrons	2.5
Fast neutrons	2.5–20*
α particles, fission fragments	20

**Depending on energy.*

for thermal neutrons to 20 for fast neutrons. For α and other multiply charged particles, w_R is also taken as 20. The values of w_R for the most commonly encountered radiations are summarized in Table 3.1.

Example 3.1

In 1 year a worker receives a γ dose of 0.01 Gy, a thermal neutron (N_s) dose of 0.002 Gy and a fast neutron dose (N_f) of 0.0002 Gy. What is his total equivalent dose? (Take the radiation weighting factor for fast neutrons as 20.)

Equivalent dose = absorbed dose × radiation weighting factor

Equivalent dose, $\gamma = 0.01 \times 1 = 0.01$ Sv

Equivalent dose, $N_s = 0.002 \times 2.5 = 0.005$ Sv

Equivalent dose, $N_f = 0.0002 \times 20 = 0.004$ Sv

Total equivalent dose = 0.019 Sv

In the remainder of the book, we generally refer to **equivalent dose** simply as dose, except where this could lead to confusion.

3.5 EFFECTIVE DOSE

A further complication is that different organs and tissues have differing sensitivities to radiation. To deal with the very common situation in which the body is not uniformly exposed, another concept is needed, which is called **effective dose**, *E*. This is obtained by summing the equivalent doses to all tissues and organs of the body multiplied by a weighting factor w_T for each tissue or organ. This is written as follows:

$$E = \sum_T H_T w_T$$

where H_T is the equivalent dose in tissue T. The basis of the organ weighting factors is discussed further in section 4.8. It should be noted that effective dose also uses units of sieverts.

3.6 SUBMULTIPLES

In terms of the levels of radiation exposure encountered in the working environment, the gray and the sievert are very large units. It is often convenient to have smaller units, and this is done by using the prefixes **milli** (one-thousandth), abbreviated to m, and **micro** (one-millionth), abbreviated to μ. Thus:

$$1\,\text{Gy} = 1000\,\text{mGy} = 1\,000\,000\,\mu\text{Gy}$$

$$1\,\text{Sv} = 1000\,\text{mSv} = 1\,000\,000\,\mu\text{Sv}$$

Example 3.2

On three successive days, a nuclear reactor operator received the following doses of γ radiation:

Day 1 95 μSv

Day 2 5 μSv

Day 3 1 mSv

What was his total dose in mSv over the 3 days?

Day	Dose	mSv
1	$95\mu Sv = \frac{95}{1000}$ mSv	0.095
2	$5\mu Sv = \frac{5}{1000}$ mSv	0.005
3	1 mSv	1.000

Total dose = 1.100 mSv

3.7 DOSE RATE

Grays and sieverts are units that express an amount of radiation which may have been received over any period of time. In controlling the radiation hazard, it is usually necessary to know the rate at which the radiation is being received. The relationship between dose, **dose rate** and time is:

$$\text{Dose} = \text{dose rate} \times \text{time}$$

Thus, if someone works in an area for 2 h and receives a dose of 4 mSv, then the dose rate in that area is 2 mSv/h. Similarly, absorbed dose rates are expressed in Gy/h.

Example 3.3

If a person is permitted to receive a total dose of 200 μSv in a week, for how many hours during that week may they work in an area in which the dose rate is 10 μSv/h?

$$\text{Dose} = \text{dose rate} \times \text{time}$$

$$\text{Time} = \frac{200\ \mu Sv}{10\ \mu Sv/h} = 20\ h$$

3.8 FLUX

It is often convenient to express a radiation field as the number of particles or photons crossing an area of 1 m^2 in 1 s. This is strictly called the **fluence rate**, but is commonly referred to as **flux** (denoted by Φ). The concept is best illustrated by a practical example.

Consider a point source which emits neutrons at the rate of Q per second (Fig. 3.3). The flux at distance r is the number of neutrons per second passing through an area of

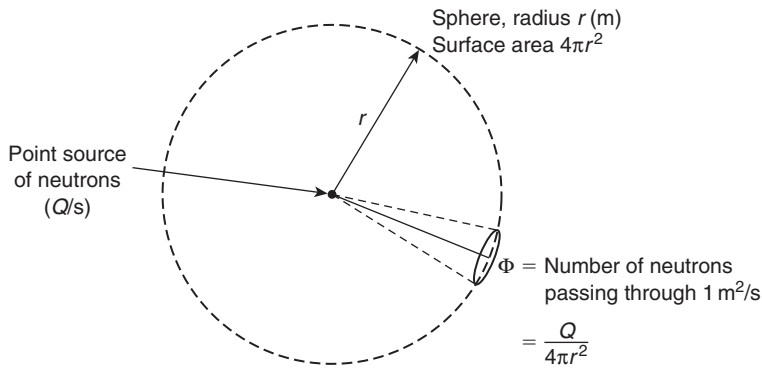


Figure 3.3 Flux from a point source.

1 m^2. Since the neutrons are being emitted uniformly in all directions, the flux at distance r is the number of neutrons emitted per second divided by the area of the sphere of radius, *r*. This area is $4\pi r^2$ and so the flux Φ is given by:

$$\Phi = \frac{Q}{4\pi r^2} \text{ neutrons per square metre per second (n/m}^2\text{/s)}$$

Note that if *r* is doubled, r^2 increases fourfold and Φ reduces fourfold. This relationship is the **inverse square law**, which will be dealt with in greater detail in section 8.31.

Example 3.4

Calculate the flux at a distance of 0.5 m from a source which emits 2×10^7 n/s.

$$\Phi = \frac{Q}{4\pi r^2}$$
$$= \frac{2 \times 10^7}{4\pi \times 0.5 \times 0.5}$$
$$= 6.4 \times 10^6 \text{ n/m}^2\text{/s}.$$

Example 3.5

Calculate the γ photon flux at 1 m from a 0.1 TBq cobalt-60 source. (Cobalt-60 emits two γ-rays per disintegration.) From Chapter 2, we know that 0.1 TBq = 10^{11} dis/s, but for ^{60}Co there are two γ-photons per disintegration. Therefore:

$$Q = 2 \times 10^{11} \text{ photons/s}$$
$$\Phi = \frac{Q}{4\pi r^2}$$
$$= \frac{2 \times 10^{11}}{4\pi \times 1^2}$$
$$= 1.6 \times 10^{10} \text{ γ photons/m}^2\text{/s}.$$

3.9 RELATIONSHIP OF UNITS

The relationship of the units which have now been introduced is illustrated in Figure 3.4. The gray describes an absorbed dose in any medium and the sievert expresses the biological effect on the human body. In radiation protection, it is clearly the biological effect of radiation that is of interest and so, whenever possible equivalent dose or effective dose should be used.

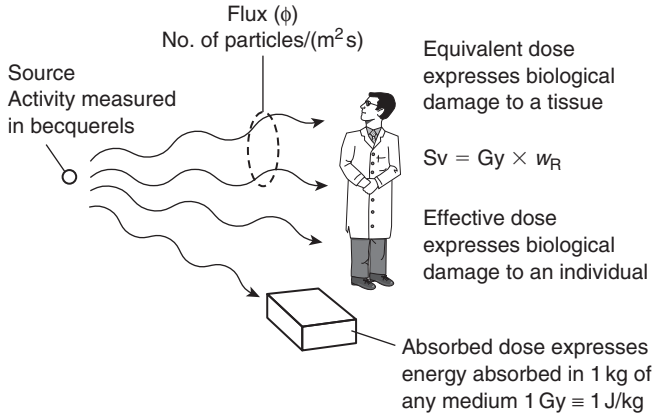


Figure 3.4 The relationship of units (with acknowledgements to Lt Cdr J. Kelly, RN).

In everyday radiation protection, the term **dose** is often loosely used to mean either of the quantities **absorbed dose** or **equivalent dose**. In the following chapters, the term dose will generally be taken to mean either **equivalent dose** or **effective dose** depending on the context.

3.10 INTERNATIONAL RADIATION SYMBOLS

The long-established and internationally agreed symbol for ionizing radiation is the trefoil symbol shown in Figure 3.5. This symbol is used on packages containing radioactive

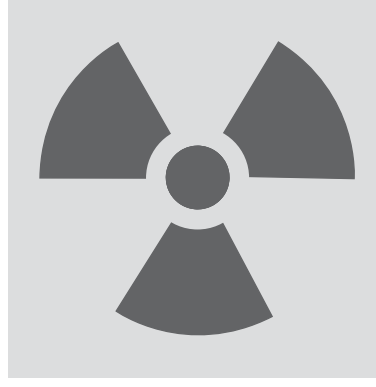


Figure 3.5 The international trefoil symbol for radiation (black on yellow).

materials and as a warning sign at the entrance to areas where there is a significant radiological hazard.

In 2007, the International Atomic Energy Agency and the International Standards Organization introduced an additional symbol for use in the special situation where very high-activity radiation sources are in use (see Fig. 3.6). The symbol is intended for use on the containers of very high-activity sources such as those in food irradiators, industrial radiography equipment and teletherapy equipment for cancer treatment (see sections 13.2.6 and 14.4). The new symbol should not be used on transport packages or at access points to radiation controlled areas.

Figure 3.6 The new symbol for very high-activity sources.

SUMMARY OF KEY POINTS

Radiation transfers energy from the source to the absorbing medium.

Ionization: removal of orbital electron – production of ion pairs.

Ionization chamber: application of electric field causes a current of ions to flow.

Absorbed dose: energy deposition in any medium by any type of ionizing radiation, 1 Gy = 1 J/kg.

Equivalent dose is obtained by multiplying the absorbed dose by the radiation weighting factor for the particular type of radiation; the equivalent dose from different types of radiation may then be added to give a measure of the overall biological effect; the unit is the sievert.

Radiation weighting factor, w_R: measure of the ability of a particular type of radiation to cause biological damage, which is related to the density of ionization. $w_R = 1$ for β, X and γ, 5 for protons, 2.5 for thermal neutrons, 2.5–20 for fast neutrons and 20 for α particles.

Effective dose: an indicator of the effects of radiation on the body as a whole when different body tissues are exposed to different levels of equivalent dose; obtained by multiplying the equivalent dose to each exposed organ by its tissue weighting factor, w_T, and then summing over all of the exposed organs.

Tissue weighting factor, w_T: a factor reflecting the radiosensitivity of a particular tissue or organ.

Dose = dose rate × time.

Flux from point source = $Q/4\pi r^2$.

REVISION QUESTIONS

1. What is ionization and how can it be used as a means of measuring radiation?
2. What is a gray?
3. Explain why the sievert is a more suitable unit in health physics than the gray.
4. Explain the concepts of equivalent dose and effective dose.
5. Calculate the neutron flux at a distance of 0.3 m from a neutron source which emits 3×10^7 n/s.
6. In 1 week an operator on a nuclear reactor works for 4 h in an area in which the γ and neutron dose rates are 5 μSv/h and 7 μSv/h, respectively, and for a further 18 h in an area in which the γ dose rate is 2 μSv/h (no neutrons). Calculate their dose for the week.
7. A worker receives an external dose in 1 year of 1 mSv (assumed to be uniform over the whole body). In addition, as a result of an intake of radioactivity, an equivalent dose of 5 mSv is delivered to the thyroid, for which the organ weighting factor is 0.04. What is their effective dose for the year?

4 Biological effects of radiation

4.1 INTRODUCTION

The interaction of ionizing radiation with the human body, arising either from **external** sources outside the body or from **internal** contamination of the body by radioactive substances, leads to biological effects which may later show up as clinical symptoms. The nature and severity of these symptoms and the time at which they appear depend on the amount of radiation absorbed and the rate at which it is received. In addition to the effects on the person receiving the dose, damage to the germ cells in the reproductive organs – the gonads – can result in heritable effects which arise in later generations.

4.2 BASIC HUMAN PHYSIOLOGY

Physiology is concerned with both the functions of the body as a whole and the component organs and systems. Some basic knowledge of physiology is necessary for an understanding of the ways in which radioactivity can enter and become distributed in the body. Humans can be regarded as machines consisting of various interrelated systems, each performing some important function. The systems that are most relevant to an understanding of the behaviour of radioactive substances that enter the body are the circulatory, respiratory and digestive systems (see Fig. 4.1).

4.2.1 The circulatory system

The circulatory system is a closed circuit of tubes around which **blood** is pumped by the action of the **heart**. Blood is the transport mechanism of the body and it circulates to almost every region, carrying nutrients (from food) and oxygen to the cells. It also picks up waste products and carbon dioxide and transfers them to the excretory organs. The heart actually has two pumps: the left side pumps the blood through the **arteries** to the tissues. Nourishment is transferred from the tissues to the cells via the tissue fluid. The blood, after passing through the tissues, returns to the right side of the heart via the **veins**. The blood is then pumped to the **lungs** where it becomes oxygenated before returning to the left side of the heart.

The blood in the arteries contains a lot of oxygen and is bright red in colour, whereas the blood returning from the tissues contains very little oxygen and is dark red with a bluish tinge. The body contains about 5 litres of blood, which circulates about once a minute. There are three types of blood cells: **red cells** (erythrocytes), **white cells** (leucocytes) and **platelets** (thrombocytes), each of which performs an essential function. The function of

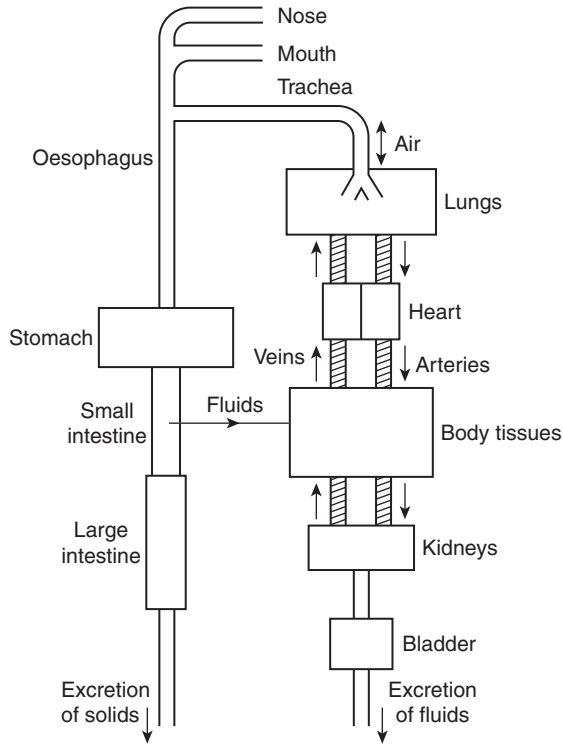


Figure 4.1 Schematic illustration of human physiology.

the **red cells** is to transport the food and oxygen required by the body, whereas the **white cells** serve as a means of defence against infection by digesting microorganisms. **Platelets** play a vital role in the formation of clots at the site of injuries.

4.2.2 The respiratory system

Respiration (or breathing) is the method by which oxygen is taken into the lungs and carbon dioxide is eliminated. Oxygen is absorbed by the blood as it passes through the lungs and is carried to the tissues as described above. The tissues produce carbon dioxide as a gaseous waste product which is carried back by the blood to the lungs and breathed out. The volume of air breathed per day is approximately 20 m^3, of which half is usually considered to be breathed during the 8 h of work.

In the process of respiration, airborne contaminants, either in the form of gaseous or particulate materials (i.e. airborne dusts), are inhaled. Gases pass freely into the lungs and enter the bloodstream to a greater or lesser extent, depending on their solubility. In the case of particulate matter, only a fraction of the inhaled material is deposited in the lungs; the remainder is either exhaled or deposited in the upper respiratory passages and subsequently swallowed. The behaviour of the material deposited in the lungs depends mainly on its solubility. Highly soluble materials are absorbed rapidly into the bloodstream, perhaps in a matter of hours, whereas insoluble material may persist in the lungs for many months. Clearly, then, the respiratory system represents a route of entry for radioactive substances

which can remain in the lungs for long periods or be transported by the bloodstream to other parts of the body.

4.2.3 The digestive system

The **digestive system** consists of the oesophagus, the stomach and the small intestine, which in turn is connected to the large intestine. Food taken in by the mouth is converted into a form suitable for the production of heat and energy, and into the molecules necessary for the growth and repair of tissues. The large molecules in the food are broken down by enzymes in the digestive tract before being absorbed into the bloodstream and passed via the liver to the tissues. The unabsorbed food, together with bacteria and cells shed from the intestine wall, is passed out as solid waste (faeces). Liquid waste (the waste products of cells dissolved in water) is excreted from the body via the kidneys and bladder as urine.

Soluble radioactive contamination, when swallowed, may pass through the walls of the digestive tract and become absorbed into the bloodstream, which carries it to all parts of the body. It is then likely to become concentrated mainly in some specific organ or tissue, which it will irradiate until it decays or is excreted. Insoluble contamination passes through the digestive tract and is excreted in the faeces. During its passage through the body, it will irradiate the tract and the large intestine.

4.3 CELL BIOLOGY

All living creatures and organisms consist of tiny structures known as cells. The basic components of a cell are the **nucleus**, a surrounding liquid known as the **cytoplasm** and a **membrane**, which forms the cell wall. Figure 4.2 shows a simplified representation of a 'typical' human cell.

The simplest description of the cell is to view the cytoplasm as the 'factory' of the cell while the nucleus contains all the information which the cell needs to carry out its function and reproduce itself. Certain structures (called organelles) within the cytoplasm break down food nutrients and convert them into energy and smaller molecules. These smaller molecules are later converted into complex molecules needed by the cell either for maintenance or for duplication.

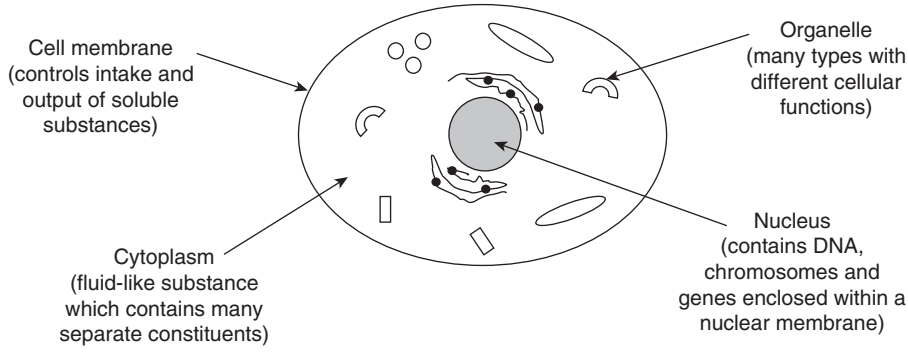


Figure 4.2 Structure of human cell (schematic).

The nucleus contains **chromosomes**, which are tiny, thread-like structures made up of genes. Human cells normally contain 46 chromosomes. The genes consist of deoxyribonucleic acid (DNA) and protein molecules, and carry the information which determines the characteristics of the daughter cell.

Cells are able to reproduce in order to compensate for cells that die. The life of different types of human cells, and hence the rate of reproduction, varies from a few hours to many years. Reproduction of cells occurs in two ways, known as mitosis and meiosis. The mitotic cells are the ordinary cells in the body and in **mitosis** the chromosomes duplicate by splitting lengthways. The original cell then divides into two new cells, each identical to the original. **Meiosis** is a special kind of division which occurs during the formation of the sexual reproduction cells, namely the sperm in the male and the ovum in the female. It occurs only once in the cell's life cycle and only in the reproductive cells. In sexual reproduction, a sperm and an ovum unite and the chromosomes combine to form a new cell containing genetic material (i.e. genes) from each of the parents. The embryo and subsequently the offspring develop from this single cell (the fertilized ovum).

4.4 THE INTERACTION OF RADIATION WITH CELLS

The basic difference between nuclear radiations and the more commonly encountered radiations such as heat and light is that the former have sufficient energy to cause ionization. In water, of which cells are largely composed, ionization can lead to molecular changes and to the formation of chemical species of a type that is damaging to the chromosome material. The damage takes the form of changes in the construction and function of the cell. In the human body, these changes may manifest themselves as clinical symptoms such as radiation sickness, cataracts or, in the longer term, cancer.

This overall process is usually considered to occur in four stages as follows:

1. The **initial physical stage**, lasting only an extremely small fraction (c. 10^{-16}) of a second in which energy is deposited in the cell and causes ionization. In water, the process may be written as:

$$H_2O \xrightarrow{\text{radiation}} H_2O^+ + e^-$$

   where H_2O^+ is the positive ion and e^- is the negative ion.

2. The **physicochemical stage**, lasting about 10^{-6} s, in which the ions interact with other water molecules, resulting in a number of new products. For example, the positive ion dissociates:

$$H_2O^+ \longrightarrow H^+ + OH$$

   The negative ion, that is the electron, attaches to a neutral water molecule, which then dissociates:

$$H_2O + e^- \longrightarrow H_2O^-$$

$$H_2O^- \longrightarrow H + OH^-$$

Thus, the products of the reactions are H^+, OH^-, H and OH. The first two ions, which are present to quite a large extent in ordinary water, take no part in subsequent reactions. The other two products, H and OH, are called free radicals, that is, they have an unpaired electron and are chemically highly reactive. Another reaction product is hydrogen peroxide, H_2O_2, which is a strong oxidizing agent formed by the reaction:

$$OH + OH \longrightarrow H_2O_2$$

3. The **chemical stage**, lasting a few seconds, in which the reaction products interact with the important organic molecules of the cell. The free radicals and oxidizing agents may attack the complex molecules which form the chromosomes. They may, for example, attach themselves to a molecule or cause links in long-chain molecules to be broken.
4. The **biological stage**, in which the timescale varies from tens of minutes to tens of years depending on the particular symptoms. The chemical changes discussed above can affect an individual cell in a number of ways. For example, they may result in:
   (a) the early death of the cell or the prevention or delay of cell division; or
   (b) a permanent modification which is passed on to daughter cells.

The effects of radiation on the human body as a whole arise from damage to individual cells, but the two types of change have quite different results. In the first case, the death or prevention of division of cells results in the depletion of the cell population within organs of the body. Below a certain level of dose (a threshold), the proportion of cells damaged will not be sufficient to affect the function of the organ and there will be no observable effect on the organ or the body as a whole. Above the threshold, effects will start to be observed and the severity of the effects will increase quite rapidly as the dose increases. This means that, within the range of variability between individuals, the relationship between the dose and the severity of the effects can be assessed with reasonable confidence. This type of effect was formerly referred to as **deterministic**, but the International Commission on Radiological Protection (ICRP; see section 6.1) has now adopted the more descriptive term **harmful tissue reaction**.

In the second case, modification of even a single cell may result, after a latency period, in a cancer in the exposed individual or, if the modification is to a reproductive cell, the damage may be transmitted to later generations and give rise to **heritable** effects. In these cases, it is the likelihood of the effect occurring that depends on the dose. This type of effect is referred to as **stochastic**, meaning 'of a random or statistical nature'.

To summarize, radiation-induced changes at the cellular level can lead to two distinct types of injury.

1. **Harmful tissue reactions** in which, above a certain threshold dose, the severity of the effects increase with increasing dose. These effects are discussed in section 4.5.
2. **Stochastic effects**, in which the probability of occurrence of the effect increases with dose. The effects include cancer induction (see section 4.6) and heritable effects in future generations (see section 4.7).

4.5 HARMFUL TISSUE REACTIONS

4.5.1 Acute radiation effects

The harmful tissue reactions that arise from acute radiation exposure (a large dose over a relatively short period of time) are those that occur within a few weeks after the receipt of the dose. The effects result from a major depletion of cell populations in a number of body organs caused by the killing of cells and the prevention or delay of cell division. The main effects are attributable to bone marrow, gastrointestinal or neuromuscular damage depending on the dose received. Acute absorbed doses above about 1 Gy (gray)give rise to nausea and vomiting. This is known as radiation sickness and it occurs a few hours after exposure as a result of damage to cells lining the intestine. Absorbed doses above about 3 Gy can lead to death, probably 10–15 days after exposures.

There is no well-defined threshold dose below which there is no risk of death from acute doses, although below about 1.5 Gy the risk of early death would be very low. Similarly, there is no well-defined point above which death is certain, but the chances of surviving an acute dose of about 8 Gy would be very low. A reasonable estimate can be made of the dose which would be lethal for 50 per cent of the exposed subjects within 60 days of exposure. This is called LD_{50}^{60} and is thought to have a value of between 3 Gy and 5 Gy for man. For doses up to about 10 Gy, death is usually caused by secondary infections that result from the depletion of white blood cells, which normally provide protection against infection. The range of doses from 3 to 10 Gy is often called the region of infection death. In this range, the chances of survival can be increased by special medical treatments, which include isolating the subject in a sterile (i.e. infection-free) environment and giving a bone marrow transfusion to stimulate white blood cell production.

For doses above about 10 Gy, survival time drops abruptly to between 3 and 5 days. It remains at about this figure until much higher doses are reached. In this region, the radiation dose causes severe depletion of the cells lining the intestine. Gross damage occurs in the lining of the intestine, followed by severe bacterial invasion. This is called the region of **gastrointestinal death**.

At much higher doses, survival times become progressively shorter. There are very few human data in this region, but from animal experiments the symptoms indicate some damage to the central nervous system and hence the region is called the region of **central nervous system death**. However, it is found that death is not instantaneous even in animals irradiated with doses in excess of 500 Gy.

Another effect which shows up soon after an acute overexposure to radiation is **erythema**, that is, reddening of the skin. In many situations the skin is subject to more radiation exposure than most other tissues. This is especially true for β-rays and low-energy X-rays. A dose of about 3 Gy of low-energy X-rays will result in erythema, and larger exposures may lead to other symptoms such as changes in pigmentation, blistering and ulceration.

The levels of exposure of workers and members of the public arising from normal operations in the nuclear energy industry, or from industrial and medical applications of radiation, are far below the levels that would induce early effects. Such high doses could be received only in the unlikely event of an accident. However, the low doses received in normal operations may cause harmful effects in the long term, and these are discussed below.

It will have been noted that, in this discussion, early effects have been considered in terms of the absorbed dose, expressed in grays, rather than as equivalent dose in sieverts

(Sv). This is really a question of definition; the radiation weighting factor, w_R, discussed in the previous chapter, and hence the concept of equivalent dose, is intended to apply only to exposures within the normal recommended limits (see Table 6.3) and should not be applied to doses at levels that could lead to early effects.

4.5.2 Late tissue reactions

Another radiation effect which may be described as a tissue reaction but which may not occur for many years is damage to the lens of the eye. This takes the form of observable opacities in the lens or, in extreme cases, visual impairment as the result of a cataract. Again, there is a threshold dose and so, by setting a dose limit for the lens of the eye, the occurrence of these effects can be prevented (see Table 6.3).

There is some evidence from animal experiments that exposure to radiation may slightly reduce the life expectation of individuals who do not exhibit any specific radiation-induced symptoms. Observations of human populations exposed at relatively high levels indicate that, if shortening of life occurs at all, it is very slight, almost certainly less than 1 year per sievert.

4.6 STOCHASTIC EFFECTS – CANCER INDUCTION

It became apparent in the early part of the twentieth century that groups of people, such as radiologists and their patients, who were exposed to relatively high levels of radiation showed a higher incidence of certain types of cancer than groups not exposed to radiation. More recently, detailed studies of the populations exposed to radiation from atomic bombs, of patients exposed to radiation therapy and of groups exposed occupationally, particularly uranium miners, have confirmed the ability of radiation to induce cancer.

Cancer is an overproliferation of cells in a body organ. It is thought that cancer may result from damage to the control system of a single cell, causing it to divide more rapidly than a normal cell. The defect is transmitted to the daughter cells, so the population of abnormal cells builds up to the detriment of the normal cells in the organ. The estimation of the increased risk of cancer is complicated by the long and variable latency period, from about 5 to 30 years or more, between exposure and the appearance of the cancer and by the fact that radiation-induced cancers are not normally distinguishable from those that arise spontaneously or as a result of other carcinogens such as tobacco smoke. The incidence of cancer in a normal population is high, with about one person in three expected to die eventually from some form of cancer. This high background makes it very difficult to establish whether any additional cases of a particular type of cancer are the result of radiation exposure, even in populations that have been exposed to relatively high levels.

At the high doses and dose rates experienced by the groups mentioned earlier, the ICRP has estimated that, averaged over a typical population of all ages, a dose of 1 Sv to each individual would result in a radiation-induced fatal cancer in about 10 per cent of the persons exposed. This is the same as saying that the average risk to an individual from a dose of 1 Sv is about 1 in 10 or 0.1. The extrapolation of this estimate to the much lower doses and dose rates normally encountered as a result of operations in the nuclear industry and elsewhere introduces further uncertainty. A very conservative approach would be to make a linear extrapolation from high to low doses. Since a dose of 1 Sv carries a risk of fatal cancer of 10 per cent, the risk from a dose of 1 mSv would be 1000 times lower, or 0.01 per

cent. However, on the basis of theoretical considerations, experiments on animals and other organisms, and limited human data, ICRP concluded that this probably overestimates the risk of radiation exposure at low doses and dose rates by a factor of between 2 and 10. This factor is referred to as the **dose and dose rate effectiveness factor (DDREF)** and, to err on the safe side, ICRP recommends using only the factor of 2. This means that the additional risk of fatal cancer imposed on an average individual by exposure to radiation at low doses and dose rates can be estimated using a risk coefficient of 0.05 per sievert (this is usually written as 5×10^{-2}/Sv). Using this coefficient, the risk of fatal cancer due to a given dose can be estimated using the relationship:

$$\text{Risk} = \text{dose (Sv)} \times \text{risk coefficient (Sv}^{-1}\text{)}$$

For a dose of 10 mSv (0.01 Sv), the risk of fatal cancer would be:

$$\text{Risk} = 0.01\ \text{Sv} \times 5 \times 10^{-2}\ \text{Sv}^{-1} = 5 \times 10^{-4}$$

In addition to fatal cancers, exposure to radiation also gives rise to cancers which are non-fatal or curable. These need to be taken into account, but clearly it would be inappropriate to give them the same weight as fatal cancers. Recognizing this, ICRP has developed the concept of detriment that allows effects of different importance to be combined to give an overall measure of the detrimental effects of radiation exposure. This is discussed further in section 4.8.

4.7 STOCHASTIC EFFECTS – HERITABLE

The heritable effects of radiation result from damage to the reproductive cells. This damage takes the form of alterations, known as **genetic mutations**, in the hereditary material of the cell.

It has already been mentioned that reproduction occurs when the ovum is fertilized by a sperm. As a result, the offspring receives a complete set of genetic material from each parent. Thus the child receives two complementary sets of genes, one from each of its parents. In general, it is found that one gene is 'dominant' and the other is 'recessive'. The dominant gene determines the particular characteristic with which it is associated.

Recessive genes are only recognized when, by chance, two of the recessive-type genes come together. A considerable number of diseases are associated with recessive genes and will therefore manifest themselves only when both parents have the same recessive genes. Spontaneous mutation accounts for the fact that an appreciable fraction of the world's population suffers from 1 of the 500 or more defects or diseases attributable to heritable effects.

Radiation can induce gene mutations which are indistinguishable from naturally occurring mutations. It should be noted in passing that heat and chemicals can also cause mutations. Mutated genes can be either dominant, in which case their effects would manifest themselves in the first generation of offspring, or recessive, when the effect would not occur in the first generation. A recessive mutation will result in an effect only if the same mutation is inherited from both parents. It is generally assumed that all mutations are harmful, although this cannot be strictly true since man has attained his present advanced

state via a series of mutations. However, this has occurred over an immense time span and the number of harmful mutations which have had to be eliminated from the species over this time is incalculably large.

Since ionizing radiation can cause an increase in the mutation rate, its use will increase the number of genetically abnormal people present in future generations. Clearly, the consequences of excessive genetic damage would be very serious indeed and strict control must be exercised over the radiation exposure of the general population.

The risks of heritable effects due to exposure of the gonads are very uncertain. Clearly, only that exposure which occurs up to the time of conception can affect the genetic characteristics of the offspring and, since the mean age of childbearing is about 30 years, only a proportion of the dose received by a typical population will be genetically harmful. As such, the ICRP estimates (ICRP *Publication 103*) that the total risk of heritable disease, up to the second generation, averaged over both sexes and all ages is about 0.2×10^{-2}/Sv. In a population of working age, because of the different age distribution, the risk is about 0.1×10^{-2}/Sv.

4.8 DETRIMENT

To assist in quantifying and combining the consequences of exposure of different organs and tissues of the body, the ICRP has developed the concept of **detriment**. This takes into account the relative risks and the average latency period of fatal cancers in different organs, an allowance for the ill health resulting from non-fatal cancers and for the risk of serious heritable effects in all future generations descended from an exposed individual. On this basis, the ICRP has provided estimates of what are termed **detriment-adjusted nominal risk coefficients** for exposure at low-dose rates, and these are shown in Table 4.1 for the population as a whole (i.e. including children) and for adults. It should be appreciated that these values are the result of calculations using data that have significant uncertainties and that, for most purposes, the use of a nominal risk coefficient of 5×10^{-2}/Sv is appropriate.

Table 4.1 Nominal risk coefficients for stochastic effects (10^{-2}/Sv)

	Cancer	Heritable effects	Total
Whole population	5.5	0.2	5.7
Adult population	4.1	0.1	4.2

In situations in which the exposure is not uniform over the body, it is necessary to know the relative contributions that individual organs make to this overall estimate of detriment, and these are shown in Table 4.2. The second column shows the probability of fatal cancer in each organ for an equivalent dose to that organ of 1 Sv. The third column gives the probability of severe heritable effects in future generations from an equivalent dose of 1 Sv to the gonads. The final column shows the relative contribution of each organ to the overall detriment, taking account of the factors discussed above.

These estimates of the relative contributions to the overall detriment from radiation exposure provide the basis for definition of the **tissue weighting factors**, w_T, used to calculate the quantity effective dose, as discussed in Chapter 3 and further explained in section 6.3.2.

Table 4.2 Relative contribution of organs to total detriment (whole population)

Organ or tissue	Probability of fatal cancer, 10^{-4} Sv^{-1}	Probability of heritable effects, 10^{-4} Sv^{-1}	Relative contribution to total detriment
Bladder	12.0	–	0.029
Bone marrow	28.0	–	0.107
Bone surface	3.2	–	0.009
Breast	33.0	–	0.139
Colon	31.3	–	0.083
Liver	28.9	–	0.046
Lung	101.5	–	0.157
Oesophagus	14.0	–	0.023
Ovary	6.0	–	0.017
Skin	2.0	–	0.007
Stomach	65.5	–	0.118
Thyroid	2.2	–	0.022
Other solid	70.5	–	0.198
Gonads	–	20	0.044
Total	398	20	1.000

SUMMARY OF KEY POINTS

Physiology: study of functions of the body as a whole and component organs and systems.

Heart pumps blood to all parts of the body via the arteries and the veins.

Blood carries food nutrients and oxygen to cells and removes waste products.

Red blood cells transport food and oxygen.

White blood cells defend the body against infection.

Platelets are vital to the formation of clots.

Respiration: method by which oxygen is taken into the lungs and carbon dioxide is eliminated.

Digestive system converts food into a form suitable for the production of heat and energy and into molecules necessary for the growth and repair of tissues.

Stages in radiation damage process:

1. Initial physical stage (c. 10^{-16} s) consisting of ionization and excitation of atoms and molecules.
2. Physicochemical stage (10^{-8}–10^{-5} s) consisting of dissociation of ions and formation of free radicals.
3. Chemical stage (a few seconds) consisting of the interaction of free radicals with other molecules in the body.

4. Biological stage (minutes to years) in which the chemical reactions show up as effects in individual cells.

Components of cell: nucleus, cytoplasm and outer membrane.

Nucleus contains chromosomes, which are thread-like structures made up of genes.

Genes carry the information which determines the characteristics of daughter cells.

Mitosis: the process by which single cells reproduce.

Meiosis: a stage in the formation of the reproductive cells – the sperm in the male and the ovum in the female.

Effects of radiation on cells: inhibition of mitosis, chromosome aberrations.

Acute effects: effects occurring within a few weeks of a very large exposure; owing to the depletion of cell populations.

Late effects: effects occurring at later times, typically some years after exposure; main effect is cancer induction.

Heritable effects may appear in descendants of exposed individuals.

Stochastic effects: the probability of occurrence depends on dose; mainly cancer and genetic effects.

Harmful tissue reactions: effects in tissues, the severity of which increases with dose, and for which a threshold may apply: mainly the early radiation effects plus certain late effects, such as cataract formation. Formerly known as deterministic effects.

Detriment: the harm from exposure to radiation, based on the probability of a stochastic effect weighted for lethality and life impairment.

Risk coefficient: the probability of a stochastic effect from a dose of 1 Sv. When the probability is weighted for the severity of the effect, it becomes a nominal risk coefficient.

REVISION QUESTIONS

1. Describe how radioactivity can be deposited in various organs of the body if it is (a) inhaled and (b) ingested (swallowed).
2. List the four stages in the radiation damage process.
3. Distinguish between harmful tissue reactions and stochastic effects of radiation.
4. What are the acute radiation effects? Discuss the severity of the effects over the dose range 1–10 Gy.
5. What is the major late effect of radiation and upon what assumptions are risk estimates based?
6. Using a nominal risk coefficient of 5×10^{-2}/Sv, calculate the risk from a dose of (a) 5 mSv and (b) 20 mSv.

5 Natural and man-made radiation

5.1 INTRODUCTION

Throughout history, man has been exposed to radiation from the environment. This natural background radiation comes from two main sources: cosmic radiation and so-called 'primordial' radiation from terrestrial sources. Note that both of these sources can lead to radioactivity within the body.

It is impossible to decide whether the natural background radiation has been harmful or beneficial to the development of the human species. It was pointed out in the previous chapter that a very small, but finite, fraction of the natural mutations in cells must be beneficial since they have contributed to the evolution of higher forms of life. Conversely, some genetic mutations lead to hereditary defects and genetic death. It is clear that these two effects have achieved some sort of balance and that life has evolved to its present state despite background radiation, or perhaps even because of it.

In addition to the natural sources of background radiation, many artificial sources of radiation have been introduced since the discovery of X-rays and radioactivity at the end of the nineteenth century, and particularly since the exploitation of the process of nuclear fission in the middle of the twentieth century. These artificial sources now add a significant contribution to the total radiation exposure of the population.

5.2 COSMIC RADIATION

Cosmic radiation reaches the Earth mainly from the Sun but also from interstellar space. It is composed of a very wide range of penetrating radiations which undergo many types of reactions with the elements they encounter in the atmosphere. The atmosphere acts as a shield and significantly reduces the amount of cosmic radiation that reaches the Earth's surface. This filtering action means that the dose rate at sea level is less than at high altitudes. For example, the mean dose rate from cosmic radiation at sea level at the equator is about 0.2 mSv/year, while the dose rate at an altitude of 3000 m is about 1 mSv/year. The average dose rate in the British Isles from cosmic radiation is about 0.33 mSv/year.

One very important radionuclide arises mainly from the interaction of neutrons in cosmic radiation with nitrogen in the upper atmosphere to form **carbon-14** as follows:

$$^{14}N\ (n, p)^{14}C$$

Carbon-14, which has a half-life of 5568 years, diffuses to the lower atmosphere, where it may become incorporated in living matter. Similarly, small concentrations of other radionuclides such as tritium (^{3}H, half-life 12.26 years), chlorine-36 (^{36}Cl, half-life 3.08×10^5 years) and calcium-41 (^{41}Ca, half-life 1.1×10^5 years) are maintained in the lower atmosphere by cosmic ray reactions. They are much less important than ^{14}C. Some of these radioisotopes, particularly ^{14}C, can be absorbed by plant life and subsequently cycled into the whole food chain.

5.3 RADIATION FROM TERRESTRIAL SOURCES

The rocks and soil of the Earth's strata contain small quantities of the radioactive elements uranium and thorium and their daughter products. The concentration of these elements varies considerably depending on the type of rock formation. In sandstone and limestone regions the concentration is much lower than in granite. Thus the dose rate depends on the geographic location. In the British Isles, the average effective dose of gamma (γ) radiation from this source is about 0.35 mSv/year. In some areas, the dose rate may be several times higher than this value.

These primordial radioisotopes are also present in low concentrations in building materials such as stone and brick used in construction and can lead to a further contribution to natural background radiation dose.

Other long-lived isotopes such as ^{48}Ca (half-life $> 7 \times 10^8$ years) and ^{50}V (half-life 4×10^{14} years) occur naturally but in very low concentrations and do not contribute significantly to human dose.

5.4 NATURALLY OCCURRING RADIOACTIVE MATERIAL (NORM)

The presence of naturally occurring radioactivity in rocks and soil also means that most natural materials are slightly radioactive. Usually the resulting radiation exposure is trivial, but there are materials that can cause significant exposure, either because they contain higher levels of naturally occurring radioactivity or because they are processed or used in such a way as to enhance the exposure. These materials are known as **NORM** (naturally occurring radioactive materials). Where materials are processed, the concentrations of the radioactivity can be increased in some of the process streams and give rise to the exposure of workers in the processing plant. In other cases, the products of processing, such as consumer products or building materials, can contain enhanced levels of activity and result in an increased radiation exposure of the general population.

In the oil and gas industries, naturally occurring **radium** and its daughter products can build up as scale in pipes and vessels. The de-scaling of these results in occupational radiation exposure and in waste streams containing radium.

In the smelting of iron ore, high concentrations of **lead-210** and **polonium-210** occur in dusts and residues. In other metal smelting applications, the use of special mineral sands containing natural **uranium** and **thorium** can lead to exposures either directly or from the enhanced concentrations in foundry slag. Another material containing levels of uranium, thorium and **potassium** that can be of radiological significance is phosphate rock. This is often used as an agricultural fertilizer. In addition, **gypsum**, which arises as a by-product of phosphate processing, is widely used in building materials.

It is the responsibility of enterprises that extract, process or use NORM to establish by appropriate surveys and assessments whether the doses are likely to be of radiological significance and, where necessary, to introduce adequate measures to ensure that exposures are kept as low as reasonably practicable.

5.5 RADIOACTIVITY IN THE BODY

The ingestion and inhalation of naturally occurring radionuclides gives rise to a dose which varies considerably depending on the location, diet and habits of the individual concerned. Potassium-40 and nuclides from the uranium and thorium series contribute most to this dose, with a minor contribution from carbon-14, which is produced by the interaction of cosmic particles with stable carbon, nitrogen and oxygen in the atmosphere.

Naturally occurring radioactivity is also taken up by plants and animals, with the result that most foodstuffs contain measurable amounts of natural radioactivity. Of ordinary foods, cereals have a high radioactive content while milk produce, fruit and vegetables have a low content. The intake of natural radioactivity varies greatly with diet and with location. The average dose in the UK from this source is about 0.25 mSv/year.

However, by far the biggest contribution to the radioactivity taken into the body comes from the inhalation of the gaseous decay products of the uranium and thorium radioactive series, namely radon and thoron.

5.5.1 Radon

Radon is a colourless and odourless gas formed from the radioactive decay of the tiny amounts of natural uranium and thorium in rocks, soils and many building materials. By far the most important isotope is ^{222}Rn, which is a member of the uranium series but, in some circumstances, ^{220}Rn (sometimes called thoron because it comes from the thorium series) can be of some significance. Radon diffuses from the ground and from building structures to give a measurable concentration in the atmosphere in the open air and, particularly, within buildings. This concentration varies significantly with geographical location depending upon the uranium content of the underlying geology. In the open air, concentrations are generally low and do not represent a significant radiological problem. Higher concentrations occur within buildings, partly as a result of diffusion from the structural materials but also because radon from the ground can enter the building. The atmospheric pressure indoors is often slightly lower than that outside, especially in the winter months, and radon gas from the ground can be drawn into the building through cracks in the floor, shrinkage gaps between the floor and the walls, as well as any service ducts. In addition, being much heavier than air, radon tends to accumulate in cellars and basements. The other major factor is the restricted ventilation within buildings. It should be noted that radon dissolves readily in water and therefore it can also be found in some natural spring or mineral waters.

Radon is the single largest contributor to background radiation dose (see Fig. 5.1). The radioactive daughter products of radon attach to dust particles which, when breathed in, irradiate the lungs and increase the risk of lung cancer. The damage is caused by alpha radiation which, despite its small range, harms cells in the sensitive lining of the lungs. Studies in the UK show that radon is responsible for 3–5 per cent of all lung cancers and, according to the US Environmental Protection Agency, radon is the second most frequent

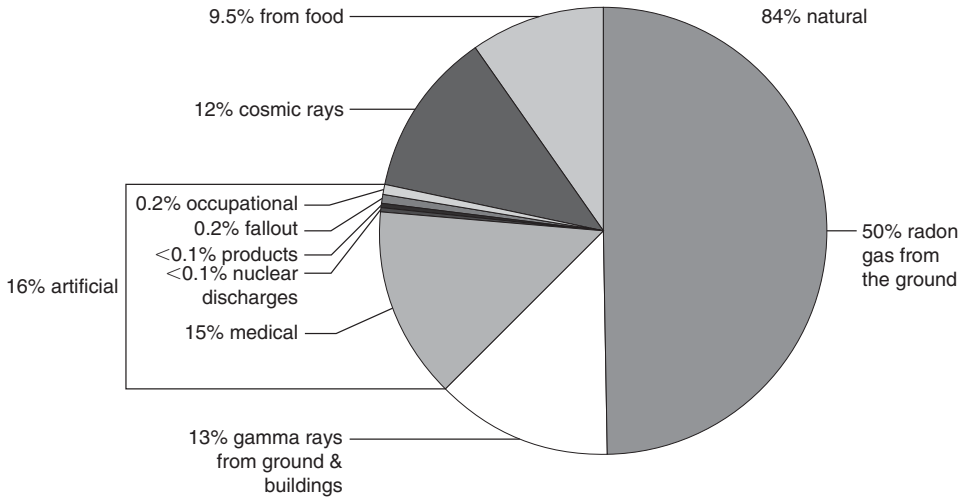


Figure 5.1 Average annual dose to the UK population. Adapted with kind permission from the Health Protection Agency leaflet, 'What is Radon?', © Health Protection Agency, Chilton, 05/2009.

cause of lung cancer, after cigarette smoking, and causes 21 000 lung cancer deaths per year in the USA.

The average annual dose to members of the UK population from this source is about 1.3 mSv/year, but studies have shown that in some dwellings, in so-called 'radon-affected' areas, the dose rate can be up to 100 times the average. As a result, there are programmes in a number of countries to identify dwellings and workplaces that have high concentrations and, where necessary, to undertake remedial work.

Radon can be 'actively' measured using specialized ionization chambers which continuously measure and record the amount of radon or its decay products in the air. However, such equipment is expensive and requires expert operation. As radon concentration inside buildings can vary significantly with time, depending on area usage and weather conditions, prolonged measurements using 'passive' radon meters often provide a more reliable and much simpler approach. A typical passive radon meter is shown in Figure 5.2, and these devices are usually placed within buildings for a 3-month period. The meter contains a sensitive plastic that registers damage tracks when exposed to α particles. The tracks can then be counted under a microscope and used to give the average radon level during the 3-month period of the measurement.

In the UK, it is recommended that an activity concentration of radon in excess of 200 Bq/m^3 in the home would necessitate some action to reduce radon, whereas in the workplace a concentration of greater than the 400 Bq/m^3 action level requires regulatory action.

The simplest approach used to reduce ingress of the radon is by sealing walls and floors and increasing the ventilation. However, in some cases it may be necessary to fit a 'radon sump' to vent the gas into the atmosphere outside the building. A sump has a pipe connecting a space under a solid floor to the outside, and a small electric fan in the pipe continually sucks the radon from under the building and expels it harmlessly to the atmosphere.

More detailed information on radon and measures to mitigate its impact can be found at the following link: http://www.ukradon.org/.

Figure 5.2 Passive Radon Metre. Adapted with kind permission from Report HPA-RRP-001 'Ionising Radiation Exposure of the UK Population: 2005 Review' by S J Watson et al (2005), © Health Protection Agency.

5.6 SUMMARY OF DOSES FROM NATURAL RADIATION

Table 5.1 gives a list of the typical average annual doses from natural radiation in the British Isles.

Local γ radiation comes from the ^{238}U and ^{232}Th series and from ^{40}K. In certain parts of the world, it is much higher than the value given in Table 5.1. For example, in the monazite sand regions of India and Brazil, the annual whole-body doses from local γ radiation can be as high as 120 mSv/year.

As mentioned earlier, exposure to cosmic radiation is quite low at ground level but increases with altitude. The annual dose to aircrew can be as much as 6 mSv.

Table 5.1 Typical average annual doses from natural radiation

Source	Dose (mSv/year)
Local γ radiation	0.35
Radon, thoron and decay products	1.30
Cosmic radiation	0.33
Ingestion of natural radioactivity	0.25
Total	~2.23

5.7 HISTORY OF MAN-MADE RADIATION EXPOSURE

The early experiences of man-made sources of radiation involved X-rays and various uses of radium. As early as 1896, a letter appeared in ***Nature*** describing the effects of repeated exposure of the hands to X-rays, and during the next 15 years many more cases were reported. These cases arose both from experiments with X-ray sets and also from their

use in various treatments. By 1911, Hesse had studied the histories of 94 cases of tumours induced in man by X-rays, of which 50 cases were among radiologists.

These studies illustrated the early forms of damage produced by X-rays and gave some indication of the longer term effects. For some types of damage, such as skin cancer, there is a latency period of between 10 and 30 years, and some radiologists observed malignant skin changes as late as 25 years after discontinuing fluoroscopic examinations. By 1922, it was estimated that more than 100 radiologists had died from radiation-induced cancer. Similarly, it has been estimated that the death rate from leukaemia among early radiologists was about nine times that among other physicians.

Other studies have shown that the average life expectancy of the pioneer radiologists was reduced by approximately 2–3 years compared with physicians in general practice. Using results from animal experiments on the relationship between shortening of life and amount of irradiation, it is estimated that the total dose received by the average radiologist from 1935 to 1958 was a few grays. The results of these studies have been used by the International Commission on Radiological Protection (ICRP) in deriving the weighting factors for irradiation of organs and tissues of the body (see section 4.8).

Experience was also gained early in the twentieth century of the effect of internal dose from various nuclides such as radium (^{226}Ra, half-life 1622 years), mesothorium (^{228}Ra, half-life 5.8 years), radiothorium (^{228}Th, half-life 1.91 years) and their daughter products. Even earlier, it had been recognized that there was a remarkably high incidence of lung cancer among the miners in the Schneeberg cobalt mines of Saxony and the Joachimsthal pitchblende mines in Bohemia. Eventually, this high rate of lung cancer was shown to be caused by radiation from the daughter products of uranium, namely ^{226}Ra, radon-222 (^{222}Rn), polonium-218 (^{218}Po) and others. These mines contain large concentrations of uranium.

During the second and third decades of the twentieth century, there were many cases of overexposure to radium. A considerable number of these arose from the use of radium as a therapeutic agent. It was administered for a large variety of diseases, ranging from arthritis to insanity.

The most serious overexposures to radium occurred in the radium dial-painting industry in the USA. Most of the persons employed were women and they had the habit of 'pointing' their paint brushes with their lips. Many of these women probably ingested megabecquerel quantities of radium. Although the damaging effects of radium were eventually established, it is not known precisely how many radium dial painters actually died from the effects of radiation damage. The study of people exposed to X-rays and radium is continuing to improve our estimates of the level of risk associated with acute and chronic exposures to radiation.

5.8 CURRENT SOURCES OF MAN-MADE RADIATION

In addition to the ever-present natural background radiation, there are several other sources of human exposure that have arisen only over the last 100 years or less. These are diagnostic radiology, therapeutic radiology, use of isotopes in medicine, radioactive waste, fall-out from nuclear weapon tests and occupational exposures to radiation.

5.8.1 Diagnostic radiology

It has been estimated that over 90 per cent of the total exposure of the population from man-made sources of ionizing radiation comes from the diagnostic use of X-rays. The

most important regions of the body in this context are the bone marrow, the colon, the gonads and the fetus. The bone marrow is the site of the primitive blood-forming cells, and so irradiation of this region can lead to the induction of leukaemia. In the colon, there is rapid cellular regeneration within the intestinal epithelium, which is particularly sensitive to radiation damage. Irradiation of the gonads is important because of the possibility of genetic damage, although recent studies suggest that gonads are far less radiosensitive than previously thought. Irradiation of pregnant women has to be controlled very strictly in order to limit the possibility of physical or mental damage to the child.

5.8.2 Therapeutic radiology

The average dose to the population from therapeutic radiology is much less than that from diagnostic radiology. Although the individual exposures used in certain treatments may be many thousands of times larger than those typically delivered in diagnostic radiography, the number of people involved (i.e. having radiotherapy treatment) is much smaller.

5.8.3 Use of radioisotopes

Radioisotopes are used in medicine to give a means of tracing the path and location of specific chemicals in the body. Since radioactive isotopes are chemically identical to stable isotopes of the same element, they will follow the same path and be concentrated to the same degree as the non-active isotopes in the body. Using suitable detectors (e.g. so-called 'gamma cameras'), the behaviour of the active, and hence by analogy of the ordinary non-active, isotopes of the element may be determined. At much higher concentrations, unsealed radioisotopes can be used for therapeutic purposes (see section 14.4).

5.8.4 Radioactive waste

The increasing use of radioisotopes and, more particularly, the development of the nuclear power industry results in an ever-growing quantity of radioactive waste. Continued dispersal of low levels of radioactive waste to the environment means that members of the general population receive radiation exposure from this source. For this reason, very strict control is exercised over the release of radioactive waste to the environment (see section 12.2). At present the contribution to the total exposure of members of the population from waste disposal is very low, about 1 μSv/year.

5.8.5 Atmospheric fall-out

In the two decades after the Second World War, several countries undertook atmospheric testing of nuclear weapons. Much of the radioactivity generated by the detonations was injected into the stratosphere (at altitudes of 10–20 km) and distributed around the world by the atmospheric circulation, gradually falling out of the atmosphere onto the surface of the earth over a period of some years. This gives rise to radiation exposure of the population, mainly through contamination of foodstuffs. The nuclides of concern in radioactive fall-out from nuclear weapons testing are similar to those arising from the operation of nuclear power stations. The two most important radionuclides are **strontium-90** (^{90}Sr, half-life 28.8 years) and **caesium-137** (^{137}Cs, half-life 30.0 years). Strontium-90 concentrates in the skeleton and caesium-137 is distributed uniformly throughout the body. Although atmospheric testing largely ceased in the 1960s, traces of these radionuclides are still measurable 50 years later because of their relatively long half-lives.

Another source of atmospheric fall-out is radioactivity released into the environment as a result of nuclear accidents, much the largest of which occurred at Chernobyl in the Ukraine in 1986. This and other accidents are discussed further in Chapter 16.

5.8.6 Occupational exposure

The dose from all occupational exposure, mainly in medicine, industry and research, is very small when averaged over the whole population. The estimated contribution to average dose in the UK is about 6 μSv/year, of which atomic energy workers contribute about 40 per cent with the remainder resulting mainly from medical exposures.

5.9 SUMMARY OF CURRENT SOURCES OF RADIATION

Table 5.2 lists the average annual doses received by members of the public in the UK from the current sources of man-made radiation.

Table 5.2 Average annual doses from man-made radiation in the UK

Source	Dose (mSv/year)
Diagnostic radiology	0.38
Therapeutic radiology	0.03
Radioactive waste	0.001
Fall-out from nuclear weapons	0.006
Occupationally exposed persons	0.006
Other sources	0.005
Approximate total	0.42

SUMMARY OF KEY POINTS

Sources of background radiation:
Cosmic radiation originating from the Sun and interstellar space. The atmosphere provides shielding.
Radiation from uranium and thorium with their daughter products in the Earth's crust.
Naturally occurring radioactive material (NORM). Material containing enhanced levels of natural radioactivity and which may need protection measures.
Radioactivity in the body. Mainly uranium and thorium plus daughters, and potassium-40.
Radon. Gaseous decay product of uranium and thorium. Inhalation of its daughter products is responsible for about half the average exposure to the population.

Man-made sources of radiation:
Medical uses of radiation and radioisotopes for diagnostic and therapeutic purposes.
Radioactivity in the environment resulting from discharges of radioactive waste, fall-out from weapons testing and nuclear accidents.
Occupational exposure from working with nuclear reactors, and from medical, industrial and military applications.

REVISION QUESTIONS

1. List the main sources of natural background radiation and discuss how these sources vary with:
   (a) altitude
   (b) geographical location.
2. Discuss the origins of radon in air and describe how its effects may be ameliorated.
3. Calculate the average dose received by members of the general population (from background radiation) in the UK over the first 30 years of their life.
4. Explain the difference between diagnostic and therapeutic radiology and comment on the contribution of each to the average dose received by members of the public.

6 The system of radiological protection

6.1 THE ROLE OF THE ICRP

The International Commission on Radiological Protection (ICRP) was established by the Second International Congress of Radiology (ICR) in 1928. Since its inception, the ICRP has been the one internationally recognized body responsible for recommending safety standards for radiation protection. It must be emphasized that the ICRP recommendations do not have any direct force of law. However, in most countries of the world, the national legislation relating to exposure to radiation is based on the recommendations of the ICRP.

The early recommendations of the ICRP were concerned with protection against X-rays and radium. Some of the earliest recommendations dealt with the length of time that a worker should be engaged on radiation work. These were:

1. not more than 7 h/day;
2. not more than 5 days/week;
3. not less than 1 month's holiday per year;
4. off days to be spent out of doors as much as possible.

The maximum permissible doses were defined very loosely.

In 1950, the ICRP extended its scope to deal with the many new problems resulting from the discovery and exploitation of nuclear fission and the birth of the nuclear industry. Since that time, there have been further revisions of the principles of radiological protection and of the dose limits recommended by ICRP. The history of the development of the dose limits for workers is shown in Table 6.1.

All of these changes have led to a decrease in the occupational exposure limits. This continuous decrease has resulted not from any positive evidence of damage to workers exposed within the earlier permissible dose levels, but rather from an increasing awareness of the uncertainties in much of the available experimental data and hence the need for caution in setting limits.

In early ICRP recommendations, the expression **tolerance dose** was used to describe the acceptable level of exposure to radiation. The term tolerance dose had the unfortunate connotation that it seemed to imply a threshold dose below which no radiation damage would occur. On the basis of a growing body of evidence that questioned the existence of a threshold dose for certain types of somatic damage, the term tolerance dose was later replaced by the term **maximum permissible dose**, which subsequently became simply **dose limit**. However, the important development was that the emphasis changed

Table 6.1 History of dose limits for workers

Dose limit	Date recommended	Comments
10 per cent of an erythema dose per year	1925	Proposed by A. Mutscheller and R. M. Sievert This corresponds to an exposure of either: c. 30 R/year from 100 kV X-rays or: c. 70 R/year from 200 kV X-rays
0.2 roentgens (R)/day or 1 R/working week	1934	Recommended by ICRP
150 mSv/year, or approximately 3 mSv/week	1950	Recommended by ICRP
50 mSv/year, or approximately 1 mSv/week	1956	Recommended by ICRP
All exposures to be kept as low as reasonably achievable; equivalent dose limit 50 mSv/year	1977	Recommended by ICRP
Limit of 20 mSv/year on effective dose	1991, 2007	Averaging over 5 years permitted subject to the requirement that the dose does not exceed 50 mSv in any one year

ICRP, International Commission on Radiological Protection.

from working to a dose limit to ensuring that all exposures are kept as low as reasonably achievable within a dose limit.

In 1991, the ICRP issued *Publication 60*, which contained new basic recommendations within an overall 'System of radiological protection'. Then, in 2007, the basic recommendations were revised in *Publication 103*, which introduced various, relatively minor, changes. The new publication also takes account of the most recent epidemiological evidence on the effects of human exposure to radiation.

6.2 THE 2007 RECOMMENDATIONS OF THE ICRP (*PUBLICATION 103*)

The system of radiological protection recommended by the ICRP recognizes three different categories of situations which can result in radiation exposure:

1. Planned exposure situations. These are situations in which a practice that will lead to radiation exposure is deliberately undertaken. An example of this is the operation of nuclear power plants, which inevitably leads to radiation exposure of workers and members of the public. Another example is the use of radiation in medical diagnosis and treatment.
2. Emergency exposure situations, where urgent action might be needed to avoid or reduce the effects of an abnormal situation. For example, in the event of a nuclear accident, volunteers may be subject to high radiation situations in order to save lives or to re-establish control and reduce the overall consequences.

3. Existing exposure situations, which apply in circumstances where people are being exposed to an existing source of radiation as a result of, for example, the after-effects of an emergency or to high levels of natural radiation.

In **planned exposure situations**, the system of protection is based on the three general and long-established principles of **justification, optimization,** and **dose and risk limitation**. However, in emergency exposure and existing exposure situations, actions are based mainly on considerations of justification and optimization, since adherence to specified limits may not always be practicable.

Justification means that whatever is done should result in sufficient benefit to offset the radiological detriment or, in other words, it should do more good than harm.

Optimization requires that for any source, the likelihood of exposure, the number of people exposed and the magnitude of individual exposures should all be kept as low as reasonably achievable (ALARA), with economic and social factors taken into account.

Dose and risk limitation means that doses and risks to individuals in any planned exposure situation are subject to limits. The dose limits recommended by the ICRP are discussed in the following section. No risk limits are specified, but it is implied that risk should be limited to levels commensurate with the level of risk associated with exposure at the relevant dose limit.

6.3 RECOMMENDED DOSE LIMITS

6.3.1 Basis of dose limits

A summary and review of information on the biological effects of ionizing radiation are included in ICRP *Publication 103* and, on the basis of the review, quantitative estimates are made of the consequences of radiation exposure, as discussed in Chapter 4. Estimates are made both for stochastic effects and for harmful tissue reactions. To reiterate:

Stochastic effects are those for which the probability of an effect occurring, rather than its severity, is regarded as a function of dose, without threshold. The most important somatic stochastic effect is the induction of cancers, for which the risk must be regarded as increasing progressively with increasing dose received, without threshold. Similarly, at the dose levels involved in radiation protection, genetic effects are regarded as being stochastic.

Harmful tissue reactions are those for which the severity of the effect varies with the dose, and for which a threshold may exist. Examples of tissue reactions are the acute radiation effects and the late effects such as cataract of the lens of the eye, damage to blood vessels and impairment of fertility. The severity of these effects varies with the size of the radiation dose received, but they are not detectable at all unless a quite high threshold dose is exceeded.

The aim of radiation protection, as stated by the ICRP, is to prevent detrimental tissue reactions and to limit the probability of stochastic effects to levels deemed to be acceptable without unduly limiting the desirable benefits associated with the exposure. This aim is achieved by:

1. setting dose limits at levels that are sufficiently low to ensure that no threshold dose is reached, even following exposure for the whole of an individual's lifetime – **prevention of harmful tissue reactions**;

2. keeping all justifiable exposures as low as is reasonably achievable, economic and social factors being taken into account, subject always to the boundary condition that the appropriate dose limits shall not be exceeded – **limitation of stochastic effects**.

The ICRP considered three levels related to the degree of tolerability of an exposure or risk. These are unacceptable, tolerable and acceptable. A limit represents a selected boundary in the region between 'unacceptable' and 'tolerable' levels. Tolerable implies that the exposure (or risk) is not welcomed, but can reasonably be tolerated, and acceptable means that the level of protection has been optimized and can be accepted without further improvement. The dose limits therefore represent the level at which continued exposure would begin to be unacceptable.

6.3.2 Recommended dose limits for workers

To limit stochastic effects, the ICRP recommends an annual effective dose limit for uniform irradiation of the whole body of 20 mSv, averaged over a period of 5 years. It is permissible to exceed 20 mSv in any one year, but the dose should never exceed 50 mSv in a year. For non-uniform irradiation of the body, weighting factors have been assigned to the various individual organs, relative to the whole body as 1.0, reflecting the harm attributable to irradiation of each organ. The sum of the weighted organ doses is known as the **effective dose**, *E*. Thus:

$$E = \sum_{\mathrm{T}} w_{\mathrm{T}} H_{\mathrm{T}}$$

where w_{T} is the weighting factor for tissue T, and H_{T} is the equivalent dose in tissue T. The annual limit on effective dose is 20 mSv, and so in any one year

$$\sum_{\mathrm{T}} w_{\mathrm{T}} H_{\mathrm{T}} \leq 20 \text{ mSv}$$

The weighting factors are given in Table 6.2.

The use of an annual effective dose limit of 20 mSv implies that, if the conditions of exposure were such that only a single tissue T were exposed, the limiting annual equivalent dose for that tissue would be:

$$\text{Dose limit}_{\mathrm{T}} = 20/w_{T} \text{ mSv}$$

For example, in the case of the lung, the weighting factor has a value of 0.12, and this implies an annual limit on equivalent dose to the lung of about 170 mSv. Similarly, for the thyroid, the weighting factor has a value of 0.04 and so the annual equivalent dose limit for the thyroid is 500 mSv. For most of the organs and tissues of the body, the 'stochastic' equivalent dose limits are lower than the threshold doses at which tissue reactions might start to occur (generally about 500 mSv, although a few tissues show higher radiosensitivities). Thus the restrictions on effective dose are sufficient to ensure the avoidance of tissue reactions in almost all tissues and organs. The exceptions are the bone surfaces, brain, salivary glands, skin, and the hands and the feet, for which an equivalent dose limit of 500 mSv in 1 year is recommended, and the lens of the eye, for which a limit of 150 mSv/year is currently recommended. However, in a subsequent statement in April 2011, the ICRP recommended a much reduced equivalent dose limit for the lens of the eye of 20 mSv in a year, averaged over defined periods of 5 years with no single year exceeding 50 mSv.

Table 6.2 Tissue weighting factors

Tissue or organ	Tissue weighting factor, w_T
Gonads	0.08
Bone marrow (red)	0.12
Colon	0.12
Lung	0.12
Stomach	0.12
Breast	0.12
Bladder	0.04
Liver	0.04
Oesophagus	0.04
Thyroid	0.04
Skin	0.01
Bone surface	0.01
Brain	0.01
Salivary glands	0.01
Remainder	0.12

The values are averages across a population of all ages and both sexes. They may be applied to workers and to members of the public.

Based on current ICRP risk factors, a 20 mSv effective (or whole body) dose leads to a fatal cancer risk of approximately 1 in 1000, which is, in an occupational situation, just tolerable.

Example 6.1

Calculate the allowable equivalent dose to the thyroid of a worker for a year in which he is exposed to non-uniform irradiation involving the whole body and the lung, as well as the thyroid. During the year he receives equivalent doses of 10 mSv to the whole body and 50 mSv to the lungs.

Using the weighting factor formula:

$$\sum_T w_T H_T \leq 20 \text{ mSv}$$

$$w_T \text{ (whole body)} \times H_T \text{ (whole body)} + w_T \text{ (lung)} \times H_T \text{ (lung)} + w_T \text{ (thyroid)} \times H_T \text{ (thyroid)} \leq 20 \text{ mSv}$$

Thus,

$$1.0 \times 10 \text{ mSv} + 0.12 \times 50 \text{ mSv} + 0.04 \times H_T \text{ (thyroid)} = 20 \text{ mSv, in the limit}$$

that is,

$$10 \text{ mSv} + 6 \text{ mSv} + 0.04\, H_T \text{ (thyroid)} = 20 \text{ mSv}$$

$$H_T \text{ (thyroid)} = \frac{20 - 16}{0.04} = 100 \text{ mSv}$$

Thus, the worker is permitted to receive up to 100 mSv equivalent dose to the thyroid during the year in question.

Example 6.2

Using the weighting factors in Table 6.2, calculate the implied limits for the gonads and the thyroid, assuming that each organ is irradiated completely in isolation.

For gonads, $w_T = 0.08$, and so

$$\text{implied annual limit} = \frac{20}{0.08} = 250 \text{ mSv}$$

For thyroid, $w_T = 0.04$, and so

$$\text{implied annual limit} = \frac{20}{0.04} = 500 \text{ mSv}$$

6.3.3 Notes on the dose limits for workers

The following points should be stressed in applying this system of dose limitation:

1. All unnecessary exposures should be avoided.
2. While it is permissible to average a worker's dose over 5 years, the effective dose should not exceed 50 mSv in any single year.
3. The Commission lays considerable emphasis on the fact that only a few workers would be expected to receive annual doses close to the recommended limit. Experience shows that in many industries the distribution of doses has often been such that the average worker has received an annual whole-body equivalent dose of about 2 mSv. Using the risk factors quoted in *Publication 103*, this implies that the average risk of death in such occupations involving radiation exposure is comparable to the average risk in other industries which are normally considered 'safe'.
4. The basis for control of the occupational exposure of women is the same as for men except that, when a pregnancy is declared, the level of protection to the fetus should be broadly similar to that for a member of the public.
5. Workplaces should be subject to classification:
   Controlled area – in which normal working conditions require workers to follow well-established procedures.
   Supervised area – where no special procedures are normally needed but exposure conditions are kept under review.

6.3.4 Recommended dose limits for individual members of the public

In *Publication 103*, the ICRP recommends an annual effective dose limit of 1 mSv for individual members of the public. However, it also recommends that, in special circumstances, a higher value of effective dose could be allowed in a single year, provided that the average over 5 years does not exceed 1 mSv/year. To prevent harmful tissue reactions, the ICRP recommends equivalent dose limits of 15 mSv/year for the lens of the eye and 50 mSv/year for the skin (Table 6.3). In their 2011 statement on the dose limit for the lens of the eye, the ICRP makes no reference to the dose limit for members of the public but the implication is that it would remain at 15 mSv/y.

Table 6.3 Recommended dose limits*

Dose limit	Occupational	Public
Effective dose	20 mSv per year, averaged over defined period of 5 years†	1 mSv in a year‡
Annual equivalent dose in:§		
lens of the eye**	20 mSv	15 mSv
skin	500 mSv	50 mSv
hands and feet	500 mSv	–

**The limits apply to the sum of the relevant doses from external exposure in the specified period and the 50-year committed dose (to age 70 years for children) from intakes in the same period. †The effective dose should not exceed 50 mSv in any single year. ‡In special circumstances, a higher value of effective dose could be allowed in a single year, provided that the average over 5 years does not exceed 1 mSv/year. §For other organs, stochastic effects are limiting and hence the dose to these other organs is controlled by the limit on effective dose. **Based on the April 2011 Statement of the ICRP on dose limits for the lens of the eye.*

6.4 PLANNED EXPOSURE SITUATIONS

This is the normal regime within which radiation protection is carried out in nuclear facilities, medical and teaching establishments, and in industry. As discussed above, the system of protection is based on the principles of justification, optimization, and dose and risk limitation.

Justification in this context means that before any new practice is introduced, a case needs to be prepared in which the benefits and detriments are assessed and showing that introduction of the practice has a positive net benefit. For example, at the highest level, the case for building a nuclear power station would need to demonstrate that the benefits to society as a whole from this form of electricity generation are sufficient to justify the radiation exposure and risk that will result. Similarly, before the introduction of any new medical diagnostic or treatment procedure involving the use of radioactive materials or other radiation sources, it needs to be shown that the benefit to patients is sufficient to justify the additional radiation exposure both to the medical staff involved and to the patients themselves. In cases such as these, the justification would normally be of a generic nature and would probably be undertaken at the national level, providing a formal case for a particular class of practice that would not then need to be justified by each individual operator or user. Justification also needs to be considered in some situations at a local or operational level. For example, in the case of a medical procedure, whether for the purpose of treatment or diagnosis, the prescribing physician needs to be sure that there is a potential benefit to the individual patient. In the context of radioactive facilities, any significant changes to plant or to operational procedures that could affect radiation exposure need to be shown to have an overall benefit.

Optimization requires that the protection measures employed in the design and operation of facilities should ensure that doses to workers should be reduced as far below the dose limit as can reasonably be achieved. This means that ways of reducing exposure should always be considered but that this should not be pursued to such an extent that the costs of the reductions are disproportionate to the benefits achieved. The ICRP recommends that in the application of this principle, the concept of a **dose or risk constraint** should be applied in a particular situation. The constraints relate to individuals but are applied to a single source. For example, if members of a population could receive exposure from several different sources, constraints would need to be applied to each source so as to ensure that

the total exposure of any individual remained within the overall dose limit. In setting constraints, account should be taken of experience in similar situations elsewhere since this could provide a benchmark of what can be achieved with good practice.

In applying the principle of dose limitation in planned exposure situations, it should always be understood that the dose limits are absolute upper limits on exposure. Experience has shown that application of the ALARA principle (i.e. optimization) usually results in doses well below the limits. In terms of dose to workers, even in relatively high dose rate situations, the average level of exposure to workers is generally 10 per cent or less of the dose limits, although there will sometimes be a few workers, such as maintenance staff, who receive doses closer to the dose limit. Similarly, it has been found that application of the ALARA principle and of dose constraints results in doses to members of the public that are well below the ICRP limits.

6.5 EMERGENCY EXPOSURE SITUATIONS

As discussed above, in emergency situations the normal dose limits do not apply and actions taken would be based on considerations of justification and optimization. This is largely achieved at any installation by the preparation of a detailed and well-rehearsed emergency plan. The emergency plan has three objectives:

1. to restrict exposures as far as is reasonably achievable and, in particular, to attempt to avoid exposures above the dose limits; this process is assisted by the use of **intervention** and **reference levels** specified by national authorities;
2. to bring the situation back under control;
3. to obtain information for assessing the causes and consequences of the incident.

In an emergency, 'informed volunteers' may receive large doses for the purpose of saving life or preventing serious injuries, or to prevent a substantial increase in the scale of the incident. Although reference levels may be laid down in emergency plans, these can only serve as guidelines. Each situation will be unique and must be assessed by those responsible for the operations at the time. The ICRP suggests that limiting effective doses to below 1 Sv should avoid serious tissue responses, and below 0.5 Sv should avoid other tissue responses. For life-saving operations, it is usually considered that whole-body doses of up to about 1 Sv could be justifiable. If the operation would require doses much in excess of this level, then the risks and possible result of the operation would have to be judged very carefully. One important consideration would be the accuracy of the information regarding the probable dose rates in the accident area; a second would be the condition of the casualties and their likelihood of survival. It may be noted that in the Fukushima event of 2011 (see section 16.4.3) a reference level of 100 mSv was set by the Japanese authorities for workers struggling to get the situation under control. This was later increased to 250 mSv as the scale of the incident escalated and the radiological conditions worsened.

Regarding exposure of members of the public following a major accident, the situation again is that it may not be possible to adhere to the normal dose limits and the requirement is for a framework to assist decisions on the implementation of countermeasures in order to avert exposure. The countermeasures that can be applied to reduce exposure of members of the public following a major release of radioactive materials include sheltering indoors, evacuation from the area and administration of stable iodine tablets (which blocks

subsequent uptake of radio-iodine in the thyroid gland). All of these carry some degree of detriment to the people concerned, and the decision to introduce countermeasures should take account of these detriments and balance them against the exposure that would be averted. This is discussed further in section 16.6.

6.6 EXISTING EXPOSURE SITUATIONS

These are situations in which there is a pre-existing source of radiation, such as a high level of natural radiation in the environment or within buildings. Another example is where there are high levels of radioactive contamination of the ground from a previous nuclear accident. Situations such as these can affect large areas and populations, and so any decisions on countermeasures must take account of their disruptive effect. Decisions are therefore based mainly on practical considerations of what is possible and on ensuring that any measures adopted result in an overall benefit. The ICRP approach is to encourage national authorities to establish reference levels that will assist the decision-making process while maintaining flexibility to take account of a wider range of factors.

SUMMARY OF KEY POINTS

International Commission on Radiological Protection (ICRP): internationally recognized body responsible for recommending a system of radiological protection.

System of radiological protection: three types of exposure situation – planned, emergency and existing.

Stochastic effects: those for which the probability of an effect occurring, rather than its severity, is regarded as a function of dose, without threshold (e.g. cancer induction or genetic effects).

Harmful tissue reactions: effects in tissues, the severity of which varies with the dose, and for which a threshold may therefore apply (e.g. acute radiation effects and late effects such as cataracts of the lens of the eye, damage to blood vessels or impairment of fertility).

Aim of ICRP recommendations: to prevent detrimental tissue reactions and to limit the probability of stochastic effects to acceptable levels.

Dose limit to avoid tissue reactions: 500 mSv in 1 year for the skin and for hands and feet, 150 mSv for the lens of the eye. For other organs, the stochastic dose limit ensures that detrimental tissue reactions will not occur.

Stochastic dose limit: an effective dose of 20 mSv/year averaged over a defined period of 5 years, with no more than 50 mSv in any one year; for non-uniform irradiation, apply the formula

$$\sum_{T} w_T H_T \leq 20 \text{ mSv}$$

with the appropriate weighting factors (w_T) for individual organs.

Controlled area: the area in which workers are required to follow well-established procedures.

Supervised area: the area in which special procedures are not normally needed but where the situation is kept under review.

Dose limits for individual members of the public: lifetime annual average effective dose limit is 1 mSv. Exceptionally, a higher dose could be allowed in a single year, provided that the average over 5 years does not exceed 1 mSv/y.

Planned exposure situations: the principles of justification, optimization and dose limitation apply.

Emergency exposure situations: adherence to dose limits may not be possible and protective actions are based on considerations of justification and optimization of protection through a system of intervention or reference levels.

Existing exposure situations: where high levels of exposure already exist owing to natural radioactivity or the effects of previous nuclear accidents; decisions on protective measures based on justification and optimization, assisted by reference levels.

Reference level: a level of dose or risk above which it may be inappropriate to allow exposures to occur and below which optimization of protection should be implemented.

REVISION QUESTIONS

1. Explain the main features of the system of radiological protection recommended by the ICRP in its *Publication 103*.
2. Explain what is meant by the terms stochastic effects and harmful tissue reactions and give two examples of each type of effect.
3. Give the annual dose limit (for workers) for each of the following: the lens of the eye; the hands; the feet.
4. Explain how the doses to various organs of the body from non-uniform irradiation are related to the whole-body limit for uniform irradiation.
5. A worker is required to work in an area where he is subjected to non-uniform irradiation, involving exposure of the whole body, the red bone marrow and the lung. During 1 year, the following equivalent doses are received:

whole body	10 mSv
lung	100 mSv
red bone marrow	150 mSv

   Calculate the committed effective dose.
6. Assuming, in turn, that each of the following organs of a worker is irradiated for the entire year in isolation, calculate the annual dose limit implied for each organ by the weighting factor formula: the thyroid, the lung and the bone surfaces.
7. State the main differences between controlled and supervised areas.
8. Explain the main considerations that should be applied to the exposure of workers and members of the public in an accident or an emergency. Why might it sometimes be permissible, following an accident, for workers to be exposed in excess of the normal operational control limits?

7 Radiation detection and measurement

7.1 GENERAL PRINCIPLES

The fact that the human body is unable to sense ionizing radiation is probably responsible for much of the general apprehension about this type of hazard. Reliance must be placed on detection devices which are based on the physical or chemical effects of radiation. These effects include:

- ionization in gases;
- ionization and excitation in certain solids;
- changes in chemical systems; and
- activation by neutrons.

Many health physics monitoring instruments use detectors based on ionization of a gas. Certain classes of crystalline solids exhibit increases in electrical conductivity and effects attributable to excitation, including scintillation, thermoluminescence and the photographic effect. Detection systems are available in which chemical changes are measured, but these are rather insensitive. A method that may be applied to neutron detection depends on the activation caused by neutron reactions.

In this chapter, the basic principles of those systems commonly used in practical health physics are described. Their applications to particular types of measurement are covered in Chapters 8, 9 and 10.

7.2 IONIZATION OF A GAS

7.2.1 Ionization chamber

It will be recalled from Chapter 3 that the absorption of radiation in a gas results in the production of ion pairs consisting of a **negative ion** (the electron) and a **positive ion**. A moderate voltage applied between two plates (**electrodes**) in close proximity causes the negative ions to be attracted to the positive electrode (**anode**) and the positive ions to the negative electrode (**cathode**). This flow of ions constitutes an electric current which is a measure of the intensity of radiation in the gas volume. The current is extremely low (typically about 10^{-12} amperes [A]) and a sensitive electronic circuit known as a direct current amplifier is used to measure it. This system is known as an **ionization chamber**, and the current measured is a mean value owing to the interaction of many charged particles or photons (Fig. 7.1).

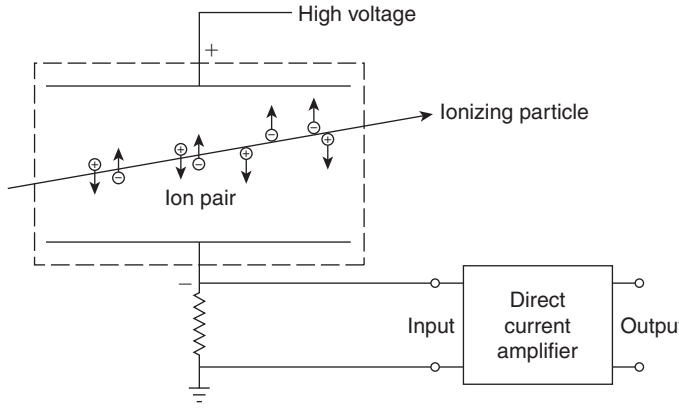


Figure 7.1 Ion chamber system.

The design of the chamber and the choice of filling gas depend on the particular application. In health physics instruments, the chamber is usually filled with air and is constructed of materials with low atomic numbers. If the instrument is required to respond to β radiation, which has a very short range in solids, the chamber must have thin walls or a thin entrance window.

7.2.2 Proportional counter

If, in an ion chamber system, the applied voltage is increased beyond a certain point, an effect known as **gas amplification** occurs. This is because the electrons produced by ionization are accelerated by the applied voltage to a sufficiently high energy to cause further ionization themselves before reaching the anode, and a cascade of ionization results (Fig. 7.2). Thus, a single ionizing particle or photon can produce a pulse of current that is large enough to be detected. Over a certain range of voltage, the size of the pulse is proportional to the amount of energy deposited by the original particle or photon, and so the system is known as a **proportional counter**. The term **counter** means that the output is a series of pulses, which may be counted by an appropriate means, rather than an average current as obtained with a direct current ionization chamber.

7.2.3 Geiger–Müller counter

If the voltage in the ionization system is increased still further, the gas amplification is so great that a single ionizing particle produces an avalanche of ionization resulting in a very large pulse of current. The size of the pulse is the same, regardless of the quantity of energy initially deposited by the particle or photon, and is governed more by the external circuit than the counter itself. The Geiger–Müller tube is very widely used in monitoring equipment because it is relatively rugged and can directly operate simple output circuits. Again, this is a counting device, but it is also possible to use a Geiger–Müller counter in a circuit which measures the average current flowing through the tube.

In practice, both proportional and Geiger–Müller counters are usually constructed in the form of a cylinder which forms the cathode, with a central thin wire which is the anode. The whole is enclosed in a glass or metal tube which is filled with a special gas mixture.

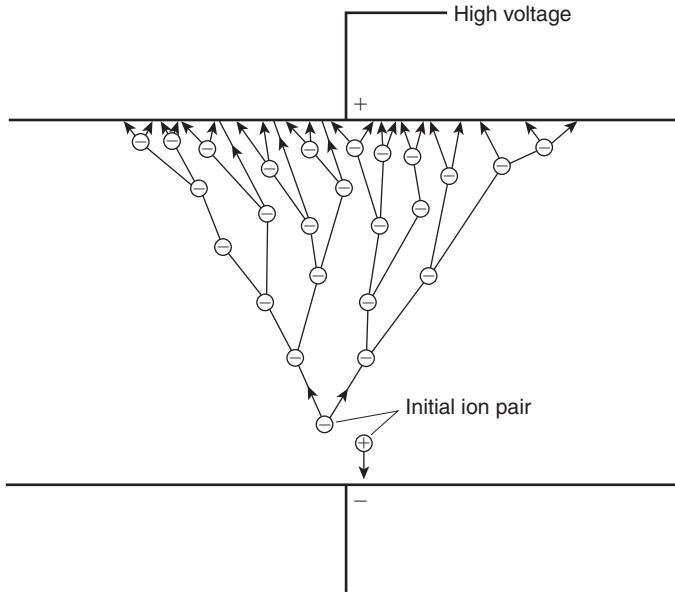


Figure 7.2 Gas amplification.

7.3 SOLID-STATE DETECTORS

7.3.1 Mechanism

The term **solid-state detectors** refers to certain classes of crystalline substances that exhibit measurable effects when exposed to ionizing radiation. In such substances, electrons exist in definite **energy bands** separated by **forbidden bands**. The highest energy band in which electrons normally exist is the **valence band**. The transfer of energy from a photon or charged particle to a valence electron may raise it from the valence band through the forbidden band into either the **exciton band** or the **conduction band**. The vacancy left by the electron is known as a hole and it is analogous to a positive ion in a gas system.

The raising of an electron to the conduction band is known as ionization, and the electron–hole pair can be compared to ion pairs in a gas. The electron and hole are independently mobile and in the presence of an electrical potential will be oppositely attracted, thus contributing to electrical conduction in the material. If an electron is raised to the exciton band, the process is excitation. In this case the electron is still bound to the hole by electrical forces and so cannot contribute to conduction. The third process that can occur is **electron trapping**. Traps are imperfections or impurity atoms in the crystal structure which cause electrons to be caught in the forbidden band. The three processes are illustrated in Figure 7.3.

The existence of the three states may be virtually permanent or they may last a very short time depending on the material and, to a great extent, the temperature. As electrons return to the valence band, the difference in energy is emitted as **fluorescent** radiation, usually a photon of visible light. In the case of trapped electrons, energy must first be provided to enable the electron to escape from the trap back into the exciton band and thence down to the valence band. The energy to release the electrons is usually provided

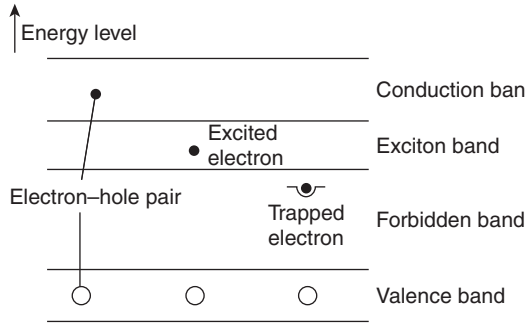


Figure 7.3 Ionization, excitation and trapping.

by raising the temperature of the substance; the light given off as a result is known as **thermoluminescence**.

The practical application of the three processes of conductivity, fluorescence and thermoluminescence is considered in more detail below. It should be mentioned that the photographic effect is also a solid-state process but is treated separately in the following text.

7.3.2 Conductivity detectors

Since changes in conductivity are caused by ionization, **solid-state conductivity detectors** are similar in some ways to gas ionization systems. A **cadmium sulphide (CdS) detector**, for example, is analogous to an ion chamber. It is operated in the mean current mode and is suitable in some applications for the measurement of the gamma (γ) dose rate. The main advantage is that it can be much smaller than a gas ionization chamber and yet have a higher sensitivity because of its much greater density.

As with gas systems, some solid-state detectors, notably germanium and silicon, operate in the pulse mode. **Germanium** has the disadvantage that it must be operated at very low temperatures. The output pulse size in both cases is proportional to the energy deposition of X-rays and γ-rays within the detector. The main application is in gamma spectrometry, in which, by analysing the size of pulses from the detector, it is possible to measure the energy of γ-rays.

7.3.3 Scintillation detectors

Scintillation detectors are based on detection of the fluorescent radiation (usually visible light) emitted when an electron returns from an excited state to the valence band. The material selected is one in which this occurs very quickly (within about 1 μs). The absorption of a 1 MeV γ-photon in a scintillation detector results typically in about 10 000 excitations and a similar number of photons of light. These **scintillations** are detected by means of a photomultiplier tube or photodiode which converts the light into electrical pulses that are then amplified. The size of pulse is proportional to the energy deposited in the crystal by the charged particle or photon. In earlier years, the most common type of scintillator used in γ-ray work was sodium iodide, usually in cylindrical crystals of about 50 mm diameter by 50 mm length. These were widely used in γ-spectrometry and had the advantages of high sensitivity and relatively low cost. They still offer advantages in some applications but have generally been supplanted by germanium detectors, which

offer better energy resolution. Zinc sulphide crystals in very thin layers are used for alpha detection and plastic scintillators are used for beta detection, again using either a photodiode or a photomultiplier to detect the scintillations. A widely used technique for the measurement of beta activity in liquid samples is **liquid scintillation counting**. Here the sample is mixed with a liquid scintillant and counted using two photomultiplier tubes and a coincidence circuit. The coincidence circuit records a pulse only when a light flash is detected by both tubes simultaneously, and this reduces the background of spurious pulses.

7.3.4 Thermoluminescence detectors

Thermoluminescence detectors use the electron trapping process. The material is selected so that electrons trapped as a result of exposure to ionizing radiation are stable at normal temperatures. If, after irradiation, the material is heated to a suitable temperature, usually about 200°C, the trapped electrons are released and return to the valence band with the emission of a light photon. Thus, if the device is heated in the dark under a photomultiplier tube, the light output can be measured, and this is proportional to the radiation dose which the detector has received. The most commonly used material is lithium fluoride, but various other materials, including calcium fluoride and lithium borate, are used in special applications.

It should be noted that, while the conductivity and scintillation methods are more suitable for measuring radiation intensity (i.e. **dose rate**), thermoluminescence detectors measure the total dose accumulated over the period of exposure.

7.4 PHOTOGRAPHIC EFFECT

Ionizing radiation affects photographic film in the same way as visible light. A photographic film consists of an emulsion of crystals (grains) of silver bromide on a transparent plastic base. The absorption of energy in a silver bromide grain, whether from light or ionizing radiation, results in the formation of a small cluster (often only a few atoms) of metallic silver. This cluster is known as a **latent image**. When the film is **developed**, this tiny amount of silver assists the conversion of all the silver in a grain from its compound form, silver bromide, into metallic silver which deposits on the plastic base material. This is an amplification process with a gain of about 10^9, which accounts for the high sensitivity of photographic emulsions. After development, the film is **fixed** or made stable by washing in a sodium thiosulphite (hypo) bath, which removes any unconverted silver bromide. If good results are to be obtained, it is important to strictly control the developer strength, temperature and processing time.

Photographic films used for radiation monitoring are usually 30×40 mm and are, of course, sealed in a light-tight packet. After processing, the film is read by passing a beam of light through it and measuring the optical density. This observed density is converted to radiation dose by means of a calibration curve obtained by exposing a number of films to known doses and plotting a dose–density curve (see Fig. 7.4).

The sensitivity of the film depends on the grain size of the emulsion. The most sensitive types give a range of dose measurement of about 50 μSv to 50 mSv.

The main advantage of photographic film is that, with the aid of special film holders incorporating filters, it enables information on the type and energy of radiation to be

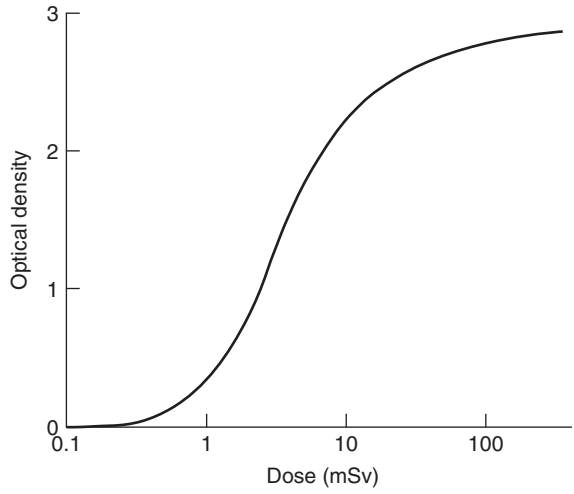


Figure 7.4 Dose–density curve.

deduced. In addition, the developed film can be stored and rescrutinized later. The most serious disadvantage is that a rapid reading cannot be obtained.

7.5 ACTIVATION EFFECT

The bombardment of most elements by neutrons produces radioactive nuclides, and measurement of the degree of activation permits an estimation of the incident neutron flux. In most cases the method is not very sensitive and its main application is in the assessment of large accidental doses.

Fast neutron measurement is often carried out using sulphur (S) discs which undergo the reaction

$$^{32}S(n, p)^{32}P \text{ (P – phosphorus)}$$

Other useful reactions for fast neutron measurement include

$$^{115}In(n, \gamma)^{116}In \text{ (In – indium)}$$

$$^{197}Au(n, \gamma)^{198}Au \text{ (Au – gold)}$$

The nuclides ^{32}P, ^{116}In and ^{198}Au are beta-emitters and are counted in a suitable system.

Another aspect of the activation effect is that a person receiving a large neutron dose (above about 0.1 Gy) would be rendered slightly radioactive and a dose estimate may be made by measurement of the induced activity. For example, activation of sodium (Na) in the body results in the production of ^{24}Na, which again is a beta-emitter.

$$^{23}Na(n, \gamma)^{24}Na$$

With moderate doses of neutrons, the decay radiation can be detected by simply holding a sensitive detector, such as a Geiger–Müller probe, against the body.

7.6 ELECTRICAL CIRCUITS

7.6.1 Types of circuit

In sections 7.2 and 7.3, various detectors were described that give an electrical signal as an output. These electrical signals are of two types: direct current and pulse. Measurement of direct current is by amplifiers, while for pulsed output, counting circuits or ratemeters are used. The important practical features of these circuits are described below.

7.6.2 Direct current amplifier

A **direct current amplifier** is a means of amplifying a very low current to a value high enough to operate a conventional ammeter or a digital display circuit. In ion chamber systems, the gain required is quite high; the current input may be about 10^{-12} A while about 10^{-6} A is necessary to drive a meter. The required gain in this case is 10^6. Amplifiers with such a high gain have a tendency to instability because of temperature fluctuations, etc. This can be reduced to a marked extent by using **negative feedback** (see Fig. 7.5). Briefly, this means that if an increased signal is fed into the input, it results in an opposite signal being fed from the output back to the input. Another important point is that when currents as low as 10^{-12} A are being measured, great care must be taken to prevent stray currents affecting the input. A very high standard of insulation and cleanliness is required between the detector output and the input to the amplifier.

The main application of direct current amplifiers is in ion chamber systems, both portable and fixed, but they are also used for solid-state detectors of the cadmium sulphide type.

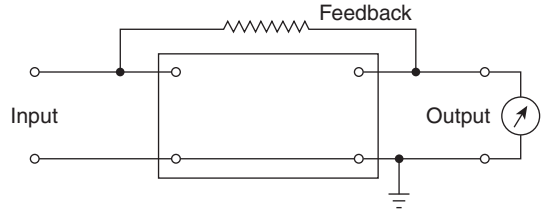


Figure 7.5 Direct current amplifier – negative feedback.

7.6.3 Pulse counting systems

A basic counting system consists of a **pulse amplifier**, a **discriminator** and a **scaler**. In addition, the system may contain a **stabilized power supply unit** to provide a supply for the detector.

The function of the pulse amplifier is to accept pulses from the detector and to amplify them to a size compatible with subsequent circuits. The size of pulses depends on the type of detector, but is typically a few microvolts for solid-state detectors, a few millivolts for proportional and scintillation counters, and up to a few volts for Geiger–Müller counters. Since most counting circuits operate on pulses of a few volts, it will be seen that the amplifier requires a gain of about 10^6 for solid-state detectors, 10^3 for proportional and scintillation counters, and can operate Geiger–Müller counting circuits directly.

In all electronic apparatus there is present, to some extent, electronic 'noise' in the form of small electrical pulses. These noise pulses are amplified with the signal pulses

from the detector and, unless precautions are taken, will be counted by the system. The function of the discriminator is to reject all pulses below a certain level, which is set by applying a discriminator bias voltage. The equipment will then record only those pulses whose amplitude exceeds the bias level. Figure 7.6 shows a train of pulses and small noise pulses being fed from an amplifier to a discriminator. If the bias level is reduced below A, noise pulses will be counted and, if it is raised above C, detector pulses will not be counted. The correct level is that indicated by B. A discriminator bias characteristic can be plotted by measuring the count rate from a detector with a range of discriminator bias settings. This is illustrated in Figure 7.7, in which the lines A, B and C correspond to those in the previous figure. Thus, if the bias voltage is below A, a very high count rate is recorded and, if it exceeds C, no counts are recorded. The correct setting is at B, at which only genuine pulses from the detector will be recorded. In addition, since the curve is almost level at this point, small variations in the bias setting will not seriously affect the count rate.

The scaler accepts pulses from the discriminator and gives a visual display of the number of pulses (counts) received over the counting period. The scaler usually

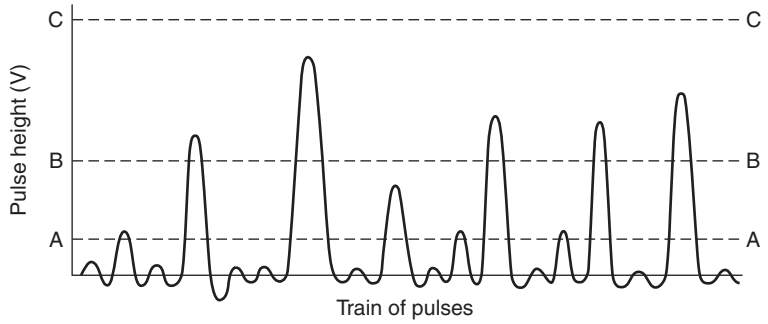


Figure 7.6 Function of a discriminator.

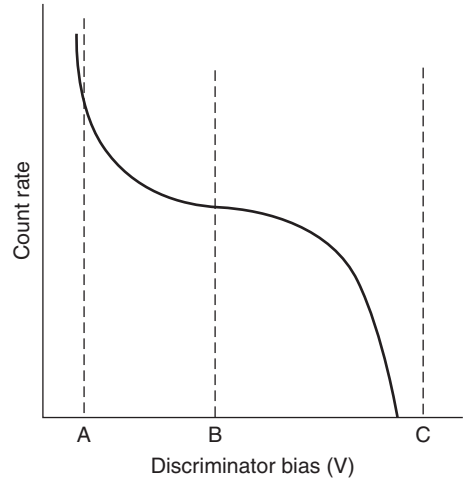


Figure 7.7 Discriminator bias characteristic.

incorporates a timing device so that once started it will count for a preset time ranging from a few seconds up to a few hours.

The counting rate from detectors is dependent on the voltage applied. To set up the equipment, a small source is placed near the detector and a series of counts are made for different detector voltages. The graph obtained by plotting these results is called a plateau because the count rate is relatively independent of the applied voltage over a certain range (see Fig. 7.8). The counter is operated at a voltage between the dotted lines (i.e. on the plateau) so that small variations in the supply voltage will not affect the response of the instrument.

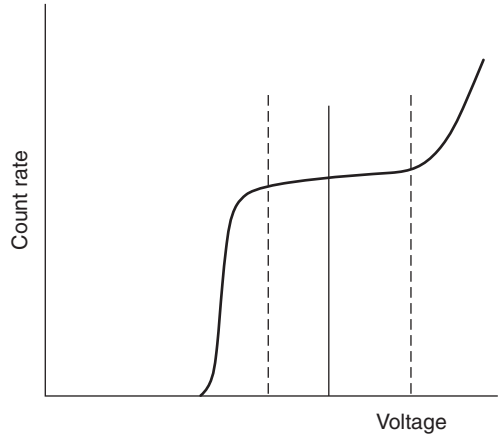


Figure 7.8 Plateau for Geiger–Müller counter.

A generalized counting system is illustrated in schematic form in Figure 7.9. The main function of this type of equipment is the measurement of radioactive samples of various types. In health physics, the samples evaluated in this way include air sample filter papers and smear and water samples. The practical aspects of sample counting are described in more detail in sections and 9.8.3 and 10.2.4.

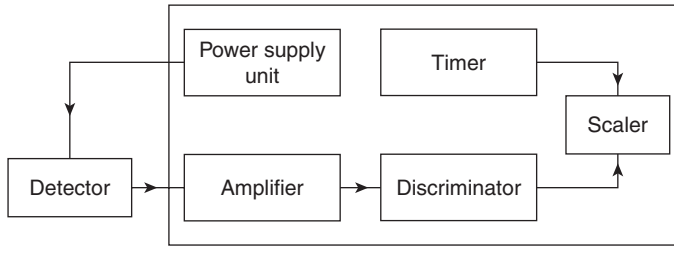


Figure 7.9 Counting equipment – schematic diagram.

7.6.4 Pulse height analyser (PHA)

When using detectors from which the output pulse height depends on the energy of the ionizing particle or photon, it is often of great help to analyse the pulses to obtain

information about the radiation spectrum. A **pulse height analyser** (**PHA**) separates the pulses into a large number of channels depending on the pulse height. Thus, if the maximum pulse height in a system is 10 V and 100 channels are available, the pulses can be segregated into channels 0.1 V wide. Any pulse smaller than 0.1 V would go into channel 1, pulses of 0.1–0.2 V into channel 2, and so on, up to pulses of 9.9–10 V, which would go into channel 100. The number of pulses going into each channel is recorded and presented on a visual display unit in such a way as to give a visual picture of the radiation spectrum. The upper line in Figure 7.10 shows a cobalt-60 (^{60}Co) γ-ray spectrum as registered by a sodium iodide (NaI) crystal. The two γ-rays of ^{60}Co really have very precise energies, but for various reasons they are 'smeared out' by the NaI detector to give the two rather broad peaks shown. This has the disadvantage that if a sample contains a mixture of radionuclides, the peaks may overlap to some extent, making it difficult to resolve the different energies. Germanium detectors offer advantages in this respect since they give very sharply defined lines and permit precise identification of the γ spectrum and hence the mixture of radionuclides. To take advantage of the high resolution of these detectors, modern PHAs have several thousands of channels.

The lower line in Figure 7.10 shows the equivalent spectrum obtained from a ^{60}Co source using a germanium detector. A disadvantage of germanium detectors is that they must be maintained at very low temperatures by means of liquid nitrogen cryostats.

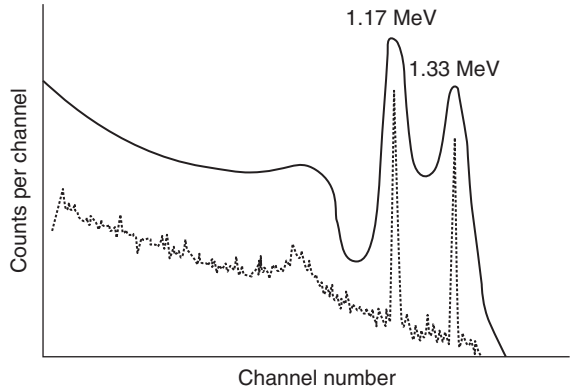


Figure 7.10 Cobalt-60 γ-ray spectra from a sodium iodide crystal (solid line) and a germanium (HPGe) detector (dotted line).

7.6.5 Ratemeter

If a detector operating in pulse mode is used in portable equipment, it is not usually practicable or desirable to use a counting system. A more convenient method of presentation is a **ratemeter**, which accepts pulses and indicates, either on a conventional meter or on a digital display, a reading related to the pulse rate.

A basic ratemeter circuit is shown in Figure 7.11. Each pulse feeds a charge into the capacitor *C*, which then slowly discharges through the resistance *R*, giving a reading on the meter. The reading on the meter or display constantly fluctuates because the arrival of a new pulse causes a sudden increase in reading followed by a slow decrease until the next pulse arrives. The degree of fluctuation can be reduced by increasing the time constant

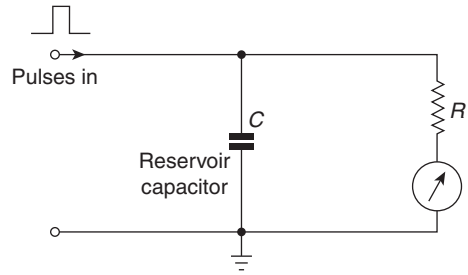


Figure 7.11 Basic ratemeter circuit.

of the circuit, which means increasing the value of *C* or *R*, or both. The time constant in seconds is obtained by multiplying *C* (farads) by *R* (ohms). Although an increase of time constant smooths out the meter reading, it also slows down the response of the circuit to sudden changes in pulse rate. After a change in pulse rate, it takes about four time constants for the ratemeter to show the new reading. For portable monitoring equipment, the time constant should not normally exceed 3 s or so.

7.7 MAINTENANCE, TESTING AND CALIBRATION OF RADIATION MONITORING INSTRUMENTATION

7.7.1 Initial testing

The manufacturer of an instrument normally carries out a detailed calibration procedure before handing it over to the user. The following items are usually investigated:

1. sensitivity of the instrument under normal working conditions;
2. energy response;
3. rate response;
4. temperature variations.

The importance of checking the energy response over a wide range of energies (usually 100 keV to several MeV) has been mentioned earlier in this chapter. Similarly, if the instrument has several scales, as is the case for most dose ratemeters, it must respond satisfactorily on all the scales. Temperature response should not be an important factor with laboratory instruments since it is always possible to select components that are practically unaffected by temperature variations.

7.7.2 Operator pre-use checks

The user of an instrument merely needs to know that the instrument is operating within specification and subsequently carries out less extensive tests to check on its performance. Most instruments have built-in checks such as battery checks and zero adjustments. The user can also test the response of the instrument with a known source as this is the parameter that is most likely to change over a period of time. The user should always carry out a background measurement prior to use. Instruments employing scintillation detectors should also be checked for light sensitivity.

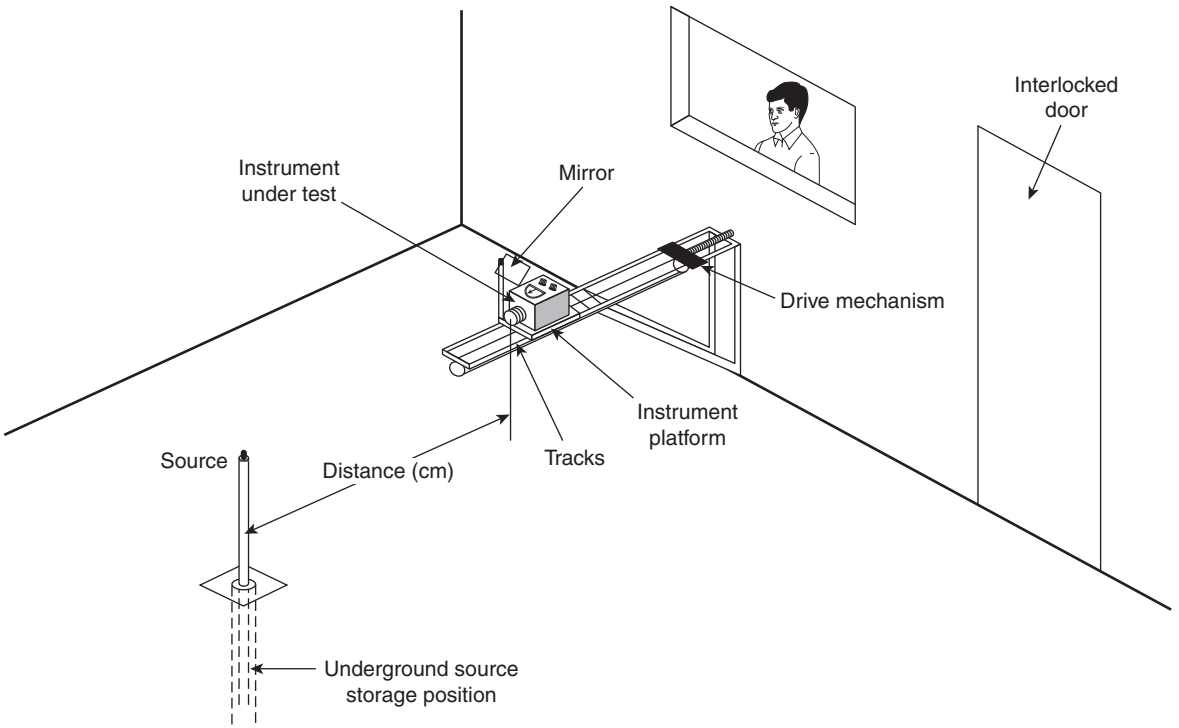


Figure 7.12 Shielded calibration room with remote operation (after Barnes and Taylor).

7.7.3 Periodic examination and testing (including calibration)

Periodically, instruments should be thoroughly examined and tested to ensure that they have not been damaged and have not lost their calibration. It is good practice to use a special calibration facility with free air conditions (that is, minimum scattering around the source) and an accurately known source to calibrate radiation dose rate-measuring instruments. One such facility is illustrated in Figure 7.12. The instrument being calibrated is moved along a guiding track and positioned remotely. Readings are obtained remotely using real-time video cameras.

Special calibration plaques with known activity concentrations of a specific radionuclide, such as ^{36}Cl or ^{241}Am, are used to calibrate contamination-monitoring instruments.

In the UK, radiation-monitoring instruments are required to be calibrated at least annually by a qualified person (QP) who possesses the necessary knowledge and expertise. A test certificate should be signed by the QP and retained by the user. Similar requirements apply in other countries.

SUMMARY OF KEY POINTS

Ionization in gases is measured by an ion chamber system with current output.

Gas amplification: ionization electrons cause further ionization if the voltage applied is high enough.

Proportional counter gives pulses of current proportional to the amount of energy deposited by original particle.

Geiger–Müller counter: each single ionizing particle causes an avalanche of electrons that gives a pulse output.

Solid-state detectors rely on ionization, excitation and electron trapping in special crystalline substances.

Conductivity detectors may be used in pulse or current mode.

Scintillation counters detect light pulses from a scintillator using a photomultiplier tube or photodiode.

Thermoluminescent detectors: energy is stored until material is heated, then light is emitted. Provides rapid read-out.

Photographic effect: blackening of film, dose–density curve. Provides a permanent dose record.

Activation effect: measures activation caused by neutrons.

Direct current: amplifier measures low currents.

Pulse counting systems consist of pulse amplifier, discriminator and scaler.

Pulse height analysers show the radiation spectrum.

Ratemeter registers the pulse rate on a meter or digital display.

Calibration of instruments: use standard source or compare with calibrated instrument.

REVISION QUESTIONS

1. Describe the operation of a Geiger–Müller counter.
2. Which class of solid-state detector is suitable for measuring a person's accumulated radiation dose? How does the detector function?
3. Write a short description of the photographic effect and discuss its advantages and disadvantages in personal dosimetry.
4. Draw a schematic diagram of a counting system and describe briefly the function of the various circuits.

8 The external radiation hazard

8.1 SOURCE OF THE HAZARD

The **external radiation hazard** arises from sources of radiation outside the body. When radioactive material actually gets inside the body, it gives rise to an **internal radiation hazard**, which requires quite different methods of control. The internal radiation hazard is discussed in Chapter 9.

The external hazard may be from beta (β), X, gamma (γ) or neutron radiation, all of which can penetrate to the sensitive organs of the body. Alpha (α) radiation is not normally regarded as an external radiation hazard as it cannot penetrate the outer layers of the skin. The external hazard is controlled by applying the three principles: time, distance and shielding.

8.2 TIME

The dose accumulated by a person working in an area with a particular dose rate is directly proportional to the amount of **time** they spend in the area. The dose can thus be controlled by limiting the time spent in the area, as defined by the equation:

$$\text{Dose} = \text{dose rate} \times \text{time}$$

Example 8.1

The annual dose limit for workers is 20 mSv per year which, assuming a 50-week working year, corresponds to 0.4 mSv, or 400 μSv per week. How many hours can a worker spend each week in an area in which the dose rate is 20 μSv/h?

$$\text{Dose} = \text{dose rate} \times \text{time}$$

$$400 = 20 \times t$$

$$\therefore t = 20 \text{ h}$$

Example 8.2

If a worker has to spend a full 40-h working week in a particular area, what is the maximum dose rate which can be allowed?

$$\text{Dose} = \text{dose rate} \times \text{time}$$

$$400 = \text{dose rate} \times 40$$

$$\therefore \text{dose rate} = 10 \text{ μSv/h}$$

Example 8.3
What would be the annual dose to a worker who spends a full working year (say 2000 h) in an area where the average dose rate is 2.5 μSv/h?

$$\text{Dose} = \text{dose rate} \times \text{time}$$

$$= 2.5 \times 2000$$

$$= 5000\ \mu\text{Sv (or 5 mSv)}$$

Example 8.4
The dose limit for individual members of the public is 1 mSv/year. What is the maximum dose rate permitted in an area that could be continuously occupied (i.e. 168 h/week) by members of the public? (Answer: ~0.12 μSv/h)

From examples 8.1–8.4 it can be seen that the dose rates that are of particular interest and that are commonly encountered in and around facilities such as nuclear reactors range from about 0.1 μSv/h up to a few tens of μSv/h. However, it should not be inferred from the examples that the only requirement is that the dose should be less than the dose limit. As discussed in section 6.2, it is required that, within the limits, doses are as low as reasonably achievable (ALARA). This involves analysing the situation to see if the sources of exposure or the time spent by workers in the area can be reduced. It also requires that means of reducing the dose rate need to be considered. The available methods are to increase the distance between the worker and the source of radiation, or to introduce some shielding material between the worker and the radiation source.

8.3 DISTANCE

8.3.1 A point source

Consider a point source of radiation which is emitting uniformly in all directions. It was shown in section 2.8 that the flux at a distance r from a point source is inversely proportional to the square of the distance r. This is due to the flux being evenly distributed over the surface area of a sphere of radius r. Since the radiation dose rate is directly related to flux, it follows that the dose rate also obeys the inverse square law. It should be noted that this is strictly true only for a point source, a point detector and negligible absorption of radiation between source and detector. The inverse square law may be written:

$$D \propto 1/r^2 \text{ or } D = k/r^2$$

$$\therefore Dr^2 = k$$

where k is a constant for a particular source.

$$\therefore D_1 r_1^2 = D_2 r_2^2$$

where D_1 = dose rate at distance r_1 from the source and D_2 = dose rate at distance r_2 from the source.

Example 8.5
The dose rate at 2 m from a particular γ source is 400 μSv/h. At what distance will it give a dose rate of 25 μSv/h?

$$D_1 r_1^2 = D_2 r_2^2$$

$$400 \times 2^2 = 25 \times r_2^2$$

$$\therefore r_2^2 = 64$$

$$\text{and } r_2 = 8 \text{ m}$$

It will be noted that doubling the distance from the source reduces the dose rate to one-quarter of its original value, trebling the distance reduces the dose rate to one-ninth and so on.

A useful expression for calculating the approximate dose rate from a γ point source is:

$$D = \frac{ME}{6r^2}$$

where D = dose rate in μSv/h, M = activity of the source in MBq, E = γ energy per disintegration in MeV and r = distance from the source in metres.

When applying this expression, care is needed in selecting the correct units. It must be emphasized that in any real situation, protection should be based on measurements of the dose rate.

Example 8.6
Calculate the approximate dose rate at a distance of 2 m from a 240-MBq cobalt-60 source. Cobalt-60 emits two γ-rays per disintegration of 1.17 MeV and 1.33 MeV.

$$D = \frac{ME}{6r^2} \text{ μSv/h}$$

$$= \frac{240 \times (1.17 + 1.33)}{6 \times 2^2}$$

$$= \frac{240 \times 2.5}{24}$$

$$= 25 \text{ μSv/h}$$

Example 8.7
Calculate the activity of a sodium-22 (^{22}Na) source which gives a dose rate of 64 μSv/h at 1 m. Assume that ^{22}Na only emits one γ-photon of energy 1.28 MeV per disintegration. (Answer: 300 MBq)

8.3.2 A line source

Another common geometry that is often encountered in practice is a line source such as a thin rod or a wire. In this case, the dose rate around the source does not follow a simple inverse square law.

Figure 8.1 shows a γ-emitting line source (e.g. an iridium-192 wire) between points 'S' and 'T'. This source is 'L' metres long and has a total activity of 'A' MBq. For simplicity, it can be assumed that the activity is uniformly distributed along the length, so that at any point along the line the activity per unit length is 'A/L' MBq per metre.

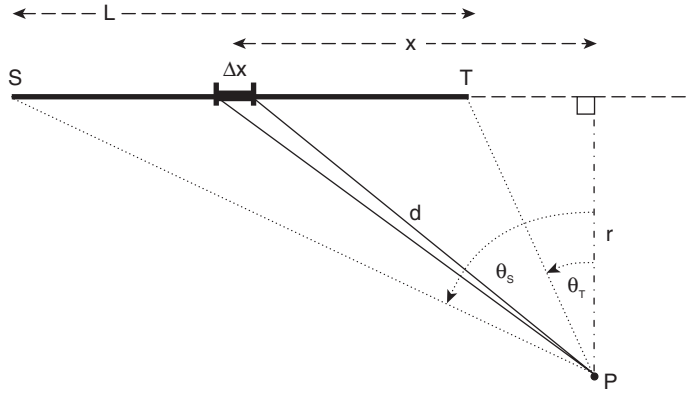


Figure 8.1 The line source.

To obtain an expression for the dose rate at some other point 'P', at a perpendicular distance of 'r' from the line, it is useful to consider the line source to be made up of lots of tiny line sources (or sections), each 'Δx' in length, joined end to end between S and T. Each tiny section has a tiny activity 'ΔA' MBq such that:

$$\Delta A = (\mathrm{A}/L)\Delta x$$

Figure 8.1 shows one section about half way along the line source. Each individual section is so tiny (i.e. so short) that it can be considered to be a point source, and so, from the section 'A point source', the dose rate at point P from this one tiny component is 'ΔD', where:

$$\Delta D = (\mathrm{A}E/6L) \times (\Delta x/d^2)$$

The total dose rate at point P is made up of tiny contributions from each tiny section. In summing (or integrating) all the dose rate contributions, from all the tiny sections from S to T, it can be shown that the total dose rate at P is:

$$D = (\mathrm{A}E/6Lr) \times (\theta_S - \theta_T)$$

where θ_S and θ_T are the angles (in radians) between point P and the two ends of the line source.

Note that at distances that are short in comparison to the length L of the source, the dose rate from the line source at point P reduces approximately as $1/r$ (and not $1/r^2$ as it was from a single point source). This indicates the 'cylindrical geometry' of a line source (as opposed to the 'spherical geometry' of a point source). Dose rate reduces with distance away from a line source; however, it does not drop off as quickly with distance as it does

from a point source. Thus, although increasing the distance from a line source is still a valid radiation protection measure, it is not as effective as it is for a point source.

8.3.3 A disc source

Using a similar 'summation' method as that described for a line source in section 8.3.2, it is possible to show that the approximate dose rate from a two-dimensional disc source is:

$$D = \frac{ME}{6a^2} \ln\left[1 + \frac{a^2}{r^2}\right]$$

where D = dose rate in μSv/h, M = total activity of the disc source in MBq, E = γ energy per disintegration in MeV, r = distance from the source in metres and a = radius of the disc source in metres.

8.4 SHIELDING

The third method of controlling the external radiation hazard is by means of shielding. Generally, this is the preferred method because it results in intrinsically safe working conditions, while reliance on distance or time of exposure may involve continuous administrative control over workers.

The amount of shielding required depends on the type of radiation, the activity of the source and the dose rate which is acceptable outside the shielding material.

Alpha particles are very easily absorbed. A thin sheet of paper is usually sufficient to stop α particles and so they never present a shielding problem.

Beta radiation is more penetrating than α radiation. In the energy range that is normally encountered (up to about 4 MeV), β radiation requires shielding of up to 10 mm of Perspex for complete absorption. The ease with which β sources may be shielded sometimes leads to the erroneous impression that they are not as dangerous as γ or neutron sources and that large open β sources may be handled directly. This is an extremely dangerous practice as, for instance, the absorbed dose rate at a distance of 3 mm from a β source of 1 MBq is about 1 Gy/h.

A significant problem, encountered when shielding against β radiation, is the emission of secondary X-rays, which result from the rapid slowing down of the β particles and which are more penetrating than the β radiation. This X radiation is known as **bremsstrahlung** and will be discussed more fully in section 13.2. The fraction of β energy reappearing as bremsstrahlung is approximately $ZE/3000$, where Z is the atomic number of the absorber and E is the maximum β energy in MeV. This means that β shields should be constructed of materials of low mass number (e.g. aluminium or Perspex) to reduce the amount of bremsstrahlung emitted.

A β source emits β particles with energies covering the complete spectrum from zero up to a characteristic maximum energy, E_{max}. The mean β energy is, in most cases, about one-third E_{max}. The penetrating power of β particles depends on their energy. This fact can be used to estimate the energy of the β radiation to aid identification of an unknown source. This will be discussed in more detail in section 10.2.2.

Gamma and X radiations are attenuated exponentially when they pass through any material. The dose rate resulting from X or γ radiation emerging from a shield can be written as:

$$D_t = D_0 e^{-\mu t}$$

where D_0 = dose rate without shielding; D_t = dose rate after passing through a shield of thickness t; and μ = linear absorption coefficient of the material of the shield.

The linear absorption coefficient μ is a function of the type of material used for the shield and also of the energy of the incident photons. It has the dimensions of length^{-1} and is usually expressed in m^{-1} or mm^{-1}.

8.4.1 Half-value layer

The half-thickness or half-value layer (HVL) for a particular shielding material is the thickness required to reduce the intensity to one-half its incident value. Writing the HVL as $t_{½}$, the previous equation becomes:

$$\frac{D_1}{D_0} = 0.5 = \exp(-\mu t_{½})$$

Taking logs to the base e:

$$\log_e 0.5 = -\mu t_{½}$$

$$\therefore -0.693 = -\mu t_{½}$$

$$\therefore t_{½} = \frac{0.693}{\mu}$$

The concept of HVL is very useful in doing rapid, approximate shielding calculations. One HVL reduces the intensity to one-half, two HVLs reduce the intensity to one-quarter, three HVLs to one-eighth and so on, as illustrated in Figure 8.2.

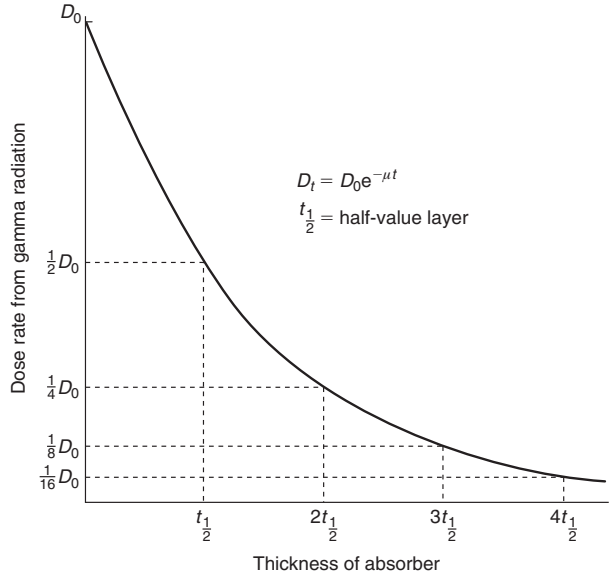


Figure 8.2 Variation of γ dose rate with absorber thickness

The value of μ, and hence $t_{1/2}$ depends on the material of the medium and on the radiation energy.

Another value sometimes used in shielding work is the **tenth-value layer**, $t_{1/10}$. By a calculation similar to that carried out above it can be shown that:

$$t_{1/10} = \frac{\log_e(10)}{\mu} = \frac{2.303}{\mu}$$

Some typical values of $t_{1/2}$ and $t_{1/10}$ for lead and water are given in Table 8.1.

Table 8.1 Approximate values of $t_{1/2}$ and $t_{1/10}$

γ Radiation [energy (MeV)]	Millimetres of lead		Millimetres of water	
	$t_{1/2}$	$t_{1/10}$	$t_{1/2}$	$t_{1/10}$
0.5	4	12.5	150	500
1.0	11	35	190	625
1.5	15	50	210	700
2.0	19	60	225	750

Example 8.8

The dose rate close to a valve is 160 μSv/h. If this is caused by cobalt-60 inside the valve, how much lead shielding must be placed around the valve to reduce the dose rate to 10 μSv/h? The HVL of lead for ^{60}Co γ radiation is 12.5 mm.

It is required to reduce the dose rate from 160 μSv/h to 10 μSv/h, that is by a factor of 16. To do this will require four HVL of lead ($2 \times 2 \times 2 \times 2 = 16$), therefore 4×12.5 mm of lead are required, that is 50 mm.

Example 8.9

A certain cobalt-60 source gives a dose rate of 40 μSv/h at 1 m. At what distance from the source must a barrier be placed if the dose rate at the barrier must not exceed 2.5 μSv/h? What thickness of lead would give the same protection at the original distance? (HVL of lead for ^{60}Co γ radiation is 12.5 mm.) (Answer: 4 m; 50 mm of lead)

Neutron shielding is complicated by the very wide range of neutron energies generally encountered. This means that any shielding equipment has to take account of a number of different energy-related reactions, the most important of which are:

1. **Elastic scatter**, in which the neutron collides with the target nucleus and 'bounces' off it in a manner similar to the collision of two billiard balls. During the collision, the neutron loses some of its initial energy and this energy is transferred to the target nucleus. All of this transferred energy appears as kinetic energy of the target nucleus. Light elements are best for slowing down neutrons by elastic scatter and so materials with a high hydrogen content (such as paraffin, water, concrete) are used.
2. **Inelastic scatter:** in this process, the incoming neutrons impart some of their energy to the scattering material and excite the target nuclei. These target

nuclei usually emit γ radiation later when they return to their ground state. The inelastic scatter process is most important for heavy nuclei.

3. **Neutron capture** reactions of many kinds: in these reactions neutrons are captured by nuclei which then de-excite by emitting another particle or photon. One very important neutron capture reaction is:

$$^{10}B(n, \alpha)^{7}Li$$

The importance of this reaction, from a shielding point of view, lies in the fact that the emitted α particle is very easily absorbed. Thus, the incorporation of boron-10 in shields means that thermal neutrons are absorbed and the resulting α particles cause no further shielding problems.

Unfortunately, the most common neutron-capture reactions lead to the emission of penetrating γ radiation, for example as in cadmium-113:

$$^{113}Cd(n, \gamma)^{114}Cd$$

Capture γ radiation is usually a limitation in shield design, and a material of high atomic number is often incorporated to absorb capture γ radiation. Nevertheless, owing to its relatively high efficiency for capturing thermal neutrons (high thermal neutron 'cross-section'), cadmium is commonly used in neutron shields.

Shielding for fast (or intermediate) neutrons might exploit elastic scatter and neutron capture by incorporating boron-10 into a material with a high hydrogen content, for example boronated polythene. The fast neutrons are first 'thermalized' by elastic scatter in the polythene and then captured by the boron.

The neutron reactions are illustrated schematically in Figure 8.3.

8.5 NEUTRON SOURCES

Nuclear fission reactors are the source of large fluxes of neutrons (see section 11.2). However, there are simpler methods for producing relatively small neutron sources. The most commonly used depend on the reaction:

$$^{9}Be(\alpha, n)^{12}C$$

A typical neutron source of this type consists of a quantity of the element beryllium mixed with an α-emitting radionuclide, usually americium-241 (^{241}Am), in a sealed capsule. For ^{241}Am–^{9}Be sources, the source strength is about 70 neutrons per second per MBq of ^{241}Am. The spectrum of neutrons emitted from an α-beryllium source is not mono-energetic but is highly peaked at energies between 3 MeV and 6 MeV; in other words, these neutron sources produce mainly fast neutrons.

Another reaction used to produce neutrons is the photoneutron process, that is the (γ, n) reaction. The most common type of photoneutron source consists of a mixture of equal volumes of antimony and beryllium, in which high-energy γ-rays from antimony-124 interact with beryllium nuclei causing the ejection of neutrons. It is worth noting that the neutrons produced by the (γ, n) process are, for most practical purposes, mono-energetic.

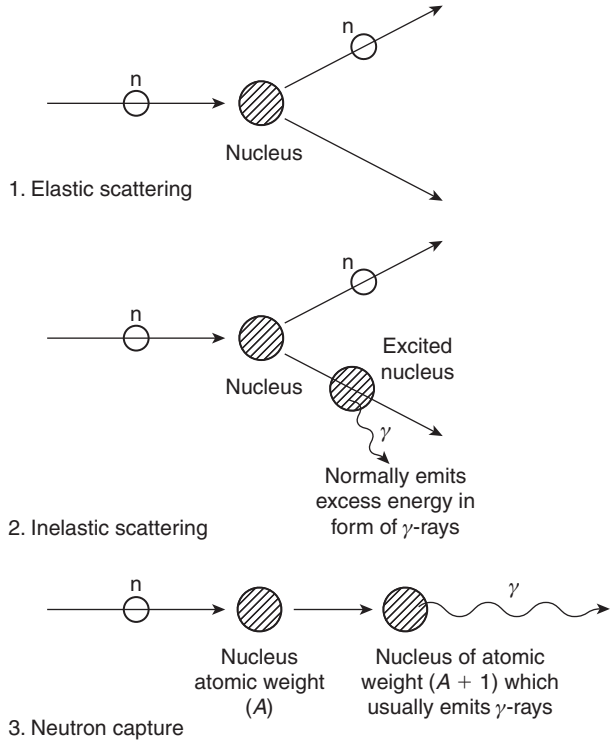


Figure 8.3 The three main neutron reactions.

To calculate the flux at a distance r from a source of strength Q, the following expression is used (see Chapter 3):

$$\Phi = \frac{Q}{4\pi r^2}$$

Example 8.10
Calculate the dose rate at 1 m from a 0.1-TBq americium–beryllium source (1 TBq of ^{241}Am–Be emits 7×10^7 n/s). Assume that 10^4 n/m^2/s is equivalent to 1 μSv/h. (Answer: 56 μSv/h)

8.6 PERSONAL DOSE CONTROL

Routine control of personal dose is based on a system of area classification. Various systems and terminologies are in use. The basic objective is to segregate areas according to the radiological hazard. In areas where the exposure is unlikely to exceed one-tenth of the occupational effective dose limit for exposed workers, that is 2 mSv/year, no special arrangements are necessary. Where workers could be exposed to a dose greater than this, but which is unlikely to exceed three-tenths of the dose limit, that is 6 mSv/year, the area would be classified as a **supervised area**. Areas in which exposure could exceed three-tenths

of the dose limit are called **controlled areas**. Workers who are likely to be exposed to a dose in excess of 6 mSv/year, which may come about through routine entry into controlled areas, are known as classified workers, and all other workers are unclassified. (In some sectors, classified and unclassified workers are referred to respectively as Category A and Category B workers.) Within controlled areas there may be regions where further demarcation is required to avoid overexposure. In some establishments these are called **restricted areas**.

A typical system of classification considers four types of area:

1. **Uncontrolled areas**, in which the time-averaged dose rate (TADR – averaged over a working day) does not exceed 1 μSv/h. Personnel can work for 40 h/week and 50 weeks/year without exceeding 2 mSv/year (one-tenth of the occupational effective dose limit in the UK).
2. **Supervised areas**, in which the dose rate (TADR) does not generally exceed 3 μSv/h and hence in which personnel will not exceed three-tenths of the dose limit. As implied by the name, these areas are subject to some form of supervision, and personnel working regularly in such areas could be subject to routine personal monitoring.
3. **Controlled areas**, in which the TADR exceeds 3 μSv/h. Personnel working regularly in controlled areas are often designated as classified (or Category A) workers and are subject to medical supervision and routine personal monitoring.
4. **Restricted areas**, in which the TADR exceeds 10 μSv/h. Access to these areas would be subject to special precautions, such as limitation of access time and the use of protective equipment and monitoring devices.

When a system of area classification is being operated, it is necessary to survey the area regularly to confirm that the classification of the area is correct and that adequate precautions are being taken. This often forms part of a risk assessment review. In controlled and restricted areas, personal dosimeters such as film badges or thermoluminescent dosimeters (TLD) must be worn to measure the accumulated dose to the worker. In addition, a direct-reading dosimeter such as an electronic personal dosimeter (EPD) is often worn to give on-the-spot control.

8.7 SURVEY MONITORING

8.7.1 Radiation survey monitoring

Radiation survey monitoring is carried out:

1. during commissioning of a facility to test the adequacy of the shielding, to show that the radiation levels are satisfactory and/or to designate areas correctly;
2. whenever changes are made that could affect radiation levels, such as changes in operations, layout or shielding arrangements;
3. routinely, during operation, to determine the working radiation levels to control the accumulated dose.

The ideal radiation survey monitor should be capable of monitoring all forms of penetrating radiation, and it should be portable, easy to use and indicate the effective dose rate. In practice, it is not possible to design a single instrument to fulfil all these requirements and so different instruments have been developed for different types of radiation.

8.7.2 X and γ radiation monitors

One type of radiation monitor measures X and γ radiation and sometimes has a facility to permit an indication (usually not very accurate) of β radiation. The actual method of detection depends on the sensitivity required. Ion chambers can only be used down to levels of a few μSv/h; below this level, the size of chamber required is too large for portable instruments. Increased sensitivity is obtained by using a Geiger–Müller tube or a scintillation detector with a circuit which measures the pulse rate.

The energy response of instruments measuring X or γ dose rate is important. In Figure 8.4, typical response curves are shown for the ion chamber, Geiger–Müller tube and scintillation detector. It is seen that the ion chamber has a relatively flat response over the energy region 0.3–10 MeV while the response curves for the Geiger–Müller tube and the scintillator tend to peak markedly at low energies. Survey monitors are often calibrated using a radium-226 source, which has an effective photon energy of 0.8 MeV. If the instrument is then used to measure radiation of different photon energy, it may seriously underestimate or overestimate the dose rate. Generally, compensating devices are incorporated into instruments using scintillation or Geiger–Müller detectors to give a relatively uniform response from about 0.1 to 3 MeV.

A typical ion chamber type survey meter is shown in Figure 8.5. This is capable of measuring β, X and γ radiation and has a number of advanced features, including automatic range selection and data-logging facilities.

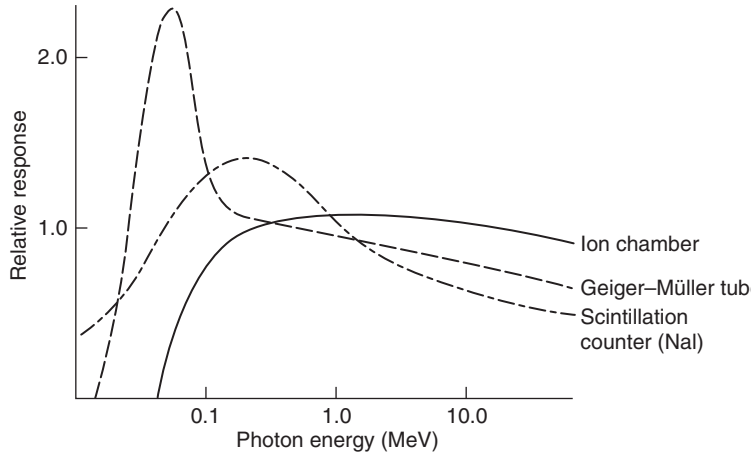


Figure 8.4 Energy response curves of various detectors.

8.7.3 Neutron monitors

Neutrons, being uncharged, do not ionize directly and so some indirect means have to be used to produce ionization. Fast neutrons are detected by causing them to interact first with a material containing a large proportion of hydrogen atoms. The neutrons 'knock on' protons in the hydrogenous material, and the ionization caused by these so-called 'recoil' protons can be detected. Fast neutron monitors often use a proportional counter with some hydrogenous material, such as polythene, incorporated in its volume. These

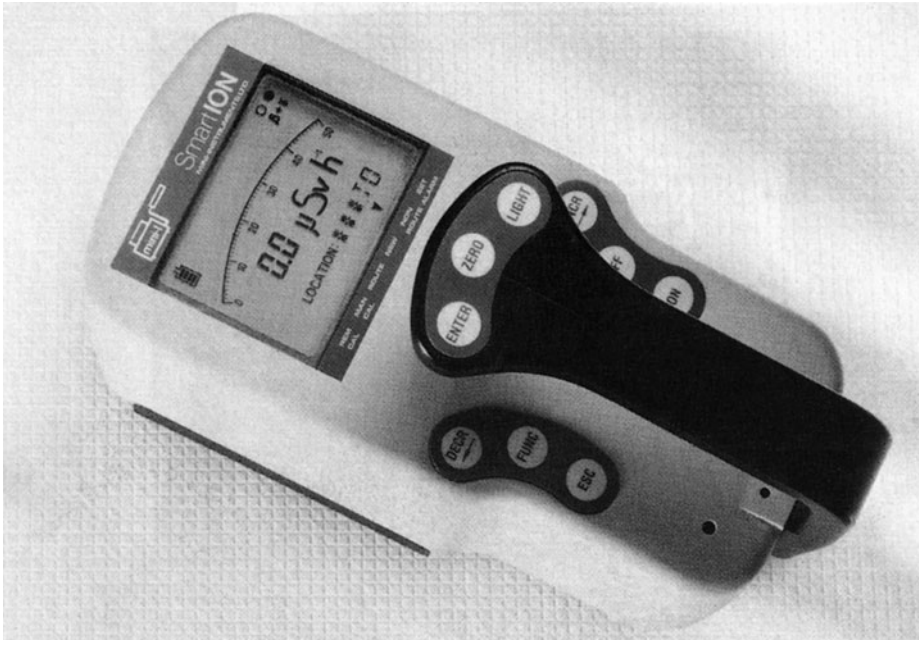


Figure 8.5 Advanced portable survey meter (courtesy of Thermo Electron Corporation).

instruments can have a very low sensitivity and it can be difficult to measure dose rates of less than about 5 μSv/h.

The most common reaction used to detect thermal neutrons is:

$$^{10}B(n, \alpha)^{7}Li$$

Boron-10 has a large cross-section for thermal neutron capture and the emitted α particles cause ionization which may then be detected. The most common thermal neutron monitors employ either an ion chamber lined with a thin layer of boron or a proportional counter filled with boron trifluoride (BF_3) gas.

The response of instruments using the boron reaction falls off rapidly above energies of a few electronvolts, whereas the instruments using the proton recoil reaction start to operate only at energies above 100 000 electronvolts (0.1 MeV). For many years, there was no instrument that could measure the intermediate energy neutrons which, it is now known, make an appreciable contribution to neutron doses around reactors. However, over the past few decades, instruments have been developed which can measure them.

The emphasis is now on instruments that measure tissue dose over a very wide range of energies from thermal up to about 15 MeV. One such instrument is illustrated in Figure 8.6. It has a cylinder of polythene which slows down fast neutrons by elastic collisions. A series of cadmium filters are arranged inside the polythene cylinder to give the correct energy response function. The thermal neutrons are detected in a proportional counter filled with helium gas.

The capture of a neutron results in the emission of a proton according to the reaction:

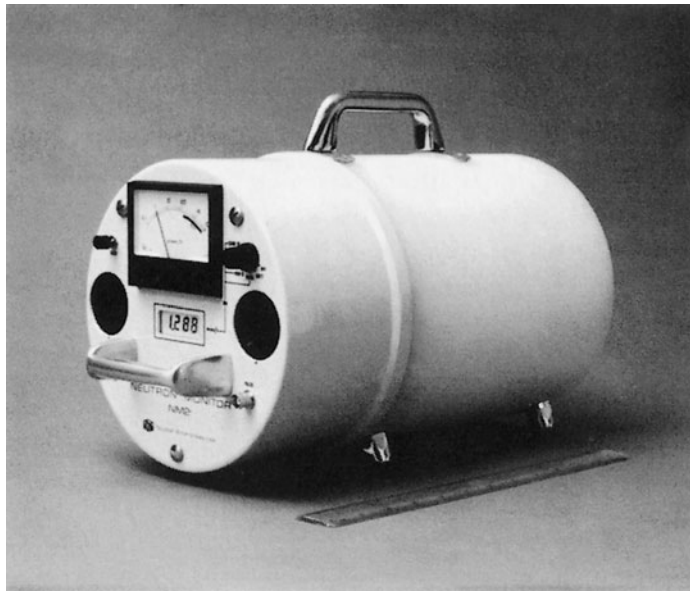


Figure 8.6 Neutron monitor (courtesy of Thermo Electron Corporation).

$$^{3}He(n, p)^{3}H$$

where ^{3}H is the isotope of hydrogen called **tritium**. Once again, ionization is caused by the proton.

This system is now generally preferred to systems using the boron-10 reaction because it is less sensitive to γ radiation.

8.8 PERSONNEL MONITORING EQUIPMENT

8.8.1 Personal dosimetry

Radiation survey monitoring is used to define radiation levels at various points in a laboratory or around a reactor. It is not an accurate method of assessing the accumulated dose received by workers in these areas because:

- It is quite likely that the dose rates will vary considerably with time, depending on the operations being carried out.
- The workers will usually move around from one radiation level to another during the course of their work.

To overcome these difficulties, it is normal practice for people working in radiation areas to wear a **personal dosimeter**. This is a device which measures the dose accumulated by the wearer and there are several types of personal dosimeter in common use.

8.8.2 The film badge

This was the traditional method of personal dosimetry and, although nowadays it has largely been replaced by other methods such as thermoluminescent dosimeters or electronic

personal dosimeters, it remains one of the accepted methods of monitoring whole-body radiation dose for the purpose of record keeping. The developed films may be stored and could, if necessary, be rescrutinized later.

The film badge in general use in the UK consists of a Kodak Personal Monitoring Type 2 film in a special holder. The Kodak Type 2 film is a double-emulsion type that has a fast emulsion on one side and a slow emulsion on the other. The degree of blackening on the developed film is determined using a densitometer and then related by calibration to the radiation exposure of the film. The use of two emulsions permits measurement over a wide range of dose. The fast emulsion enables γ-ray doses in the range of 50 μSv to about 50 mSv to be measured. If a dose in excess of about 50 mSv has been received, part of the fast emulsion is stripped from the film and the slow emulsion then permits measurements up to 10 Sv.

The holder is illustrated in Figure 8.7. It incorporates several filters so that β, γ, X and thermal neutron doses can be measured. Basically, β dose is measured in the open window area, γ dose is measured under the lead and thermal neutron dose is measured by taking the difference between the lead + cadmium and the lead + tin filters. Thermal neutrons interact with cadmium via an (n, γ) reaction and the resulting γ rays give additional blackening under the cadmium filter. The 300 mg/cm^2 and 50 mg/cm^2 plastic filters permit energy corrections to be applied to the β and low-energy γ doses. The strip of lead around

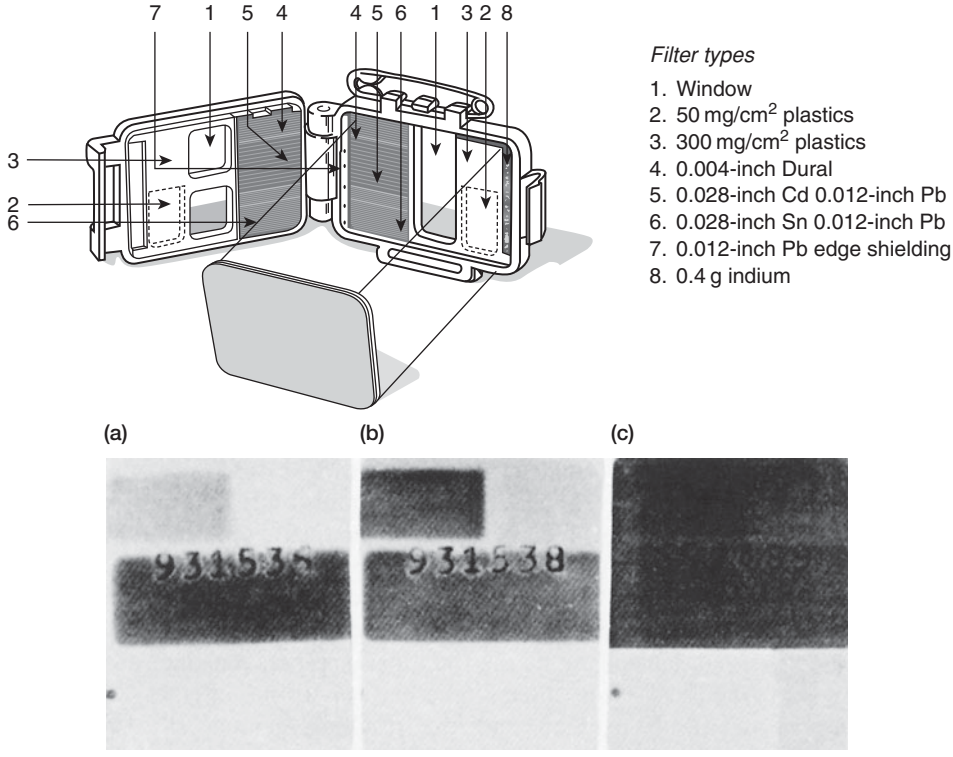


Figure 8.7 The film badge. (After Heard and Jones, courtesy of the United Kingdom Atomic Energy Authority.) Patterns are typical of (a) a high-energy β exposure, (b) an X-ray exposure and (c) a γ exposure.

the edge of the holder is to minimize edge effects at the boundary of the various filters. The indium foil is an 'exposure indicator' for criticality accidents. A dose above 10 mSv of thermal neutrons will activate the indium sufficiently to permit measurement using a Geiger–Müller tube.

8.8.3 Thermoluminescent dosimeters

These materials offer an accurate and stable means of measuring dose over the short and long term and find applications both as whole-body and extremity monitors. The action of ionizing radiation on thermoluminescent materials and the method of reading have been described in section 7.3.4. One of the disadvantages of this technique is that the process of reading the dose destroys the information, so that, unlike the film badge, the dosimeter can be read only once. The TLD system also provides less information about the quality of the radiation.

Two materials currently in use for occupational dose monitoring are lithium fluoride and calcium fluoride. The latter is very sensitive but has a poor energy response. Lithium fluoride is less sensitive but its energy response is excellent. In a practical dosimeter system, the thermoluminescent material is usually in the form of a thin disc.

Another material that can be used to assess medical X-ray doses is lithium borate which has a 'tissue-equivalent' effective atomic number and as such does not affect the image quality. Also, calcium sulphate has a very high sensitivity and can be used for environmental dose measurements.

Many establishments are now using TLD systems as the primary method of personal monitoring. This is because they are particularly suitable for automatic linking to computerized dose-recording systems. Even in establishments in which the film badge is the main method of personal monitoring, TLD systems are used to provide a convenient method of short-term dose control, particularly for dose to the extremities, for example the fingers.

An example of a practical TLD system is shown in Figure 8.8.

8.8.3 Optically stimulated luminescence dosimeters

These are similar to TLD devices; however, the detector material is aluminium oxide (Al_2O_3) crystals dispersed in a plastic wafer. Instead of using heat (as in the TLD), the amount of radiation exposure is measured with a green light from either a laser or a light-emitting diode. The amount of blue light emitted from the crystals is proportional to the radiation exposure.

A typical optically stimulated luminescence (OSL) badge is shown in Figure 8.9.

8.8.4 Electronic personal dosimeter

Various types of electronic personal dosimeters (EPDs) have been available for some decades, mostly based on miniature Geiger–Müller tubes, providing a direct display of either dose rate or accumulated dose. In many cases, the dosimeters incorporated alarm features to give warning of a high dose rate or of reaching a pre-determined accumulated dose. Their main application has been to provide an 'on-the-job' method of dose measurement and control.

A new generation of electronic dosimeters has become available using solid-state detectors and taking advantage of developments in information technology. By means of inbuilt microprocessors and memory, they can be programmed to perform a variety of

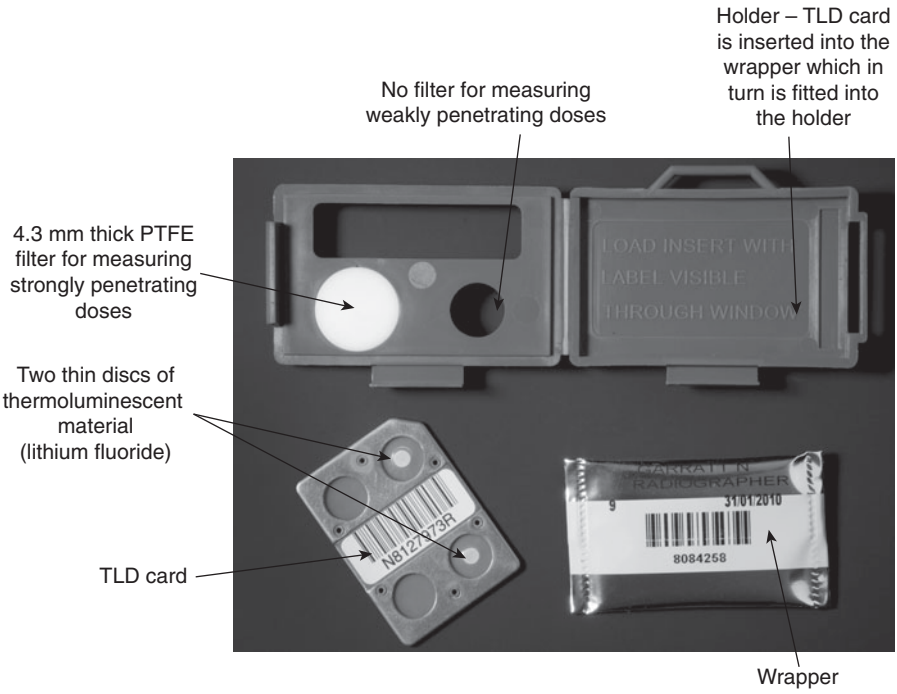


Figure 8.8 Thermoluminescent dosimeter (courtesy of the Health Protection Agency). PTFE, polytetrafluoroethylene; TLD, thermoluminescent dosimeter.

Figure 8.9 Optically stimulated luminescence (OSL) dosimeter (courtesy of Landauer, Inc., reproduced with permission).

functions, such as logging doses for a specific task or for a shift, or storing information on the characteristics of the radiation field. With gate entry facilities they can be used as a form of security pass, giving recorded access to controlled areas. The devices therefore offer the advantage of combining the operational dose control and the long-term legal

dose measurement functions. An example of a modern electronic dosimeter is shown in Figure 8.10.

The disadvantages, compared with film dosimeters or TLD, are that they are relatively costly initially, bulky and are potentially susceptible to electromagnetic fields.

8.8.5 Fast neutron dosimeter

The fast neutron dosimeter consists of a nuclear emulsion film sealed in a moisture-proof sachet and worn in a holder similar to a film badge. The holder contains lead, boron and plastic filters to screen the emulsion from body-scattered neutron radiation. The fast neutrons interact with the base material of the film and cause recoil protons to be ejected. These protons create ionization tracks in the emulsion which show up when the film is developed. The tracks are counted under a microscope and the number of tracks per square centimetre is a measure of the neutron dose.

Track plates have a range of 1 mSv to 1 Sv. Their chief drawbacks are that they are expensive to evaluate and have a threshold energy of about 0.5 MeV for the detection of neutrons. In addition, they are sensitive to γ radiation and a dose of about 100 mSv of γ makes the counting of tracks impossible.

8.8.6 Criticality locket

A criticality locket is worn in addition to the film badge or TLD whenever fissile material is handled (e.g. fuel-element manufacturing and processing plants, fuel-element cooling ponds and reactors). It is designed to measure the very high doses which could be experienced during a criticality accident. The criticality locket contains components which are activated by neutrons of different energies. The reactions utilized in the dosimeter are:

Figure 8.10 Commercially available electronic personal dosimeter (courtesy of Thermo Electron Corporation).

$^{32}S(n, p)^{32}P$ (S, sulphur; P, phosphorus)

$^{197}Au(n, \gamma)^{198}Au$ (Au, gold)

$^{115}In(n, n)^{115m}In$ (In, indium)

All of the activated components are β emitters and can be counted in a suitable shielded detection system. From the counts obtained, the fast, intermediate and thermal neutron dose can be estimated.

8.9 RADIATION RECORDS

The purpose of personal dosimetry is to ensure that workers' exposed to ionizing radiation are kept within the dose limits specified in the appropriate legislation or code of practice. In most countries, personal monitoring devices other than those for short-term dose control must be provided by an approved laboratory. An approved dosimetry laboratory has the following duties, to:

1. issue film badges, TLDs and any other personal dosimeters (such as fast-neutron track plates) which may be necessary;
2. process and assess these on return;
3. issue dose reports and maintain dose records.

Some of the larger employers of workers who wear personal dosimetry devices have established their own processing laboratories. Other users rely on specialist organizations such as, in the UK, the Health Protection Agency.

For legal purposes, both film badges, TLDs and OSL are approved methods of personal dosimetry in the UK. They are normally processed once a month and the results are recorded in the worker's personal dose record. In addition to a monthly dose report, a quarterly summary is normally issued in respect of each employee which summarizes the total radiation dose accumulated over the calendar quarter, year and working life.

As discussed in section 8.8.4, electronic dosimeters are developing to the extent that they are suitable for use within an approved dosimetry system, although they are not yet widely used for this purpose.

SUMMARY OF KEY POINTS

External radiation hazard arises from radioactive materials outside the body.

Control of external hazard: time, distance and shielding.

Time: Dose = dose rate × time.

Distance: Inverse square law $D_1 r_1^2 = D_2 r_2^2$.

Dose rate from a γ source

$$D = \frac{ME}{6r^2}\ \mu\text{Sv/h}$$

(M in MBq, E in MeV, r in metres).

Shielding: Alpha particles very easily absorbed.

Beta radiation. Use low Z materials to reduce bremsstrahlung.

Gamma radiation is attenuated exponentially,

$D_t = D_0 e^{-\mu t}$

$\text{HVL} = t_{\frac{1}{2}} = 0.693/\mu$

Neutron shielding: elastic scatter, inelastic scatter and neutron capture.

Neutron sources depend on either (α, n) or (γ, n) reactions.

Area classification: uncontrolled, supervised, controlled and restricted areas.

X and γ monitors use ion chambers, Geiger–Müller tubes or scintillation detectors.

Neutron monitors ideally cover the energy range from thermal up to about 15 MeV, use the reaction $^3\text{He(n, p)}^3\text{H}$ to give good γ rejection.

Personal dosimeters:

Film badge, special film with two emulsions to cover range from 50 μSv to 10 Sv, in a holder with filters which allow measurement of β, γ, X and thermal neutron dose.

Thermoluminescent dosimeters store the radiation energy which can later be released by heating. The light output is measured using a photomultiplier tube, the electrical output of which is a measure of the radiation dose.

Optically stimulated luminescent dosimeters are similar to TLDs but are read with the intense light from either a laser or a light-emitting diode.

Electronic personal dosimeter: based on solid-state detectors and provides both short-term and long-term measurement capability with direct readout.

Fast neutron track plate: special film in a holder, fast neutrons eject recoil protons which cause developable tracks in the emulsion. Main disadvantage is that track plates are expensive to evaluate.

Criticality locket: worn when handling fissile material; the various components are activated by neutrons of different energy and can be counted in a β castle.

REVISION QUESTIONS

1. What are the three methods by which the external radiation hazard is controlled?
2. To carry out a certain process, a worker has to work in an area which has an average dose rate of 10 μSv/h. How many hours per week can he or she work in this area if 100 μSv/week is not to be exceeded? To what level must the dose rate be reduced to allow work in the area for 40 h/week?
3. Calculate the equivalent dose rate at a distance of 1 m from a 540-MBq cobalt-60 source. At what distance will the equivalent dose rate be 25 μSv/h?
4. The dose rate at a distance of 1 m from a certain γ source is 360 μSv/h. At what distance from the source is the dose rate 10 μSv/h?
5. Calculate the approximate dose rate at a distance of 2 m from a 3000-MBq γ source which emits one 1.6-MeV γ photon per disintegration.
6. Discuss the problem of detecting neutrons and show how it is overcome in modern neutron monitors.
7. List the main types of personal dosimeter and discuss the relative advantages and disadvantages of the film badge, the TLD and the electronic dosimeter.

9 The internal radiation hazard

9.1 UNCONTAINED RADIOACTIVITY

When a radioactive material is enclosed inside some form of sealed container, it may give rise to an external radiation hazard to personnel working in its vicinity. Conversely, when radioactive material is not contained in any way it also constitutes a potential **internal radiation hazard**. Uncontained radioactive material is generally referred to as **contamination**.

Quite small quantities of radioactive material which represent an insignificant external hazard can give rise to appreciable dose rates if they come into contact with, or get inside, the body. Once the radioactive substance is taken into the body, it will continue to irradiate the body until either the radioactivity has decayed or the body has excreted the substance. The rate of decay of the radioactivity depends on its half-life, which can vary from a small fraction of a second to many thousands of years. The rate of excretion of the substance from the body depends mainly on its chemical characteristics, and it may happen in a period of a few days or it may take much longer, perhaps many years. Thus, when a radioactive substance enters the body, it may irradiate it for only a few days or for a much longer period that may extend to many years in the case of certain nuclides.

9.2 ROUTES OF ENTRY

There are three ways in which contamination can enter the body. These are:

1. **inhalation** of airborne contamination;
2. **ingestion**, that is entry through the mouth; and
3. **entry through the skin**, or through a contaminated wound.

It should be noted that contamination can also result in direct irradiation of the skin.

When airborne contamination is inhaled, a proportion of the radioactivity is deposited in the lungs and the respiratory tract and the remainder is exhaled. Some of the deposited material is eliminated from the lungs quite quickly and is swallowed. The material remaining in the lungs may then be absorbed at a greater or lesser rate into the bloodstream, be transported to other body organs and eventually be excreted. The fraction initially deposited and its rate of clearance depends on many factors, such as the physical and chemical form of the material and the metabolism of the person involved. Generally, materials in an insoluble form can remain in the lungs and continue to irradiate them for many years (depending on the half-life), while more soluble forms of material will be cleared

from the lungs into other body organs, which will in turn be subject to irradiation until the material decays or is excreted. Similarly, when contamination is ingested, the amount of it passing through the wall of the digestive tract into the body fluids and into body organs depends on the nature of the contamination and on metabolic and physiological factors.

There are wide variations in the characteristics of human beings and, in order to provide a consistent basis for radiation dose calculations, the International Commission on Radiological Protection (ICRP) has defined a set of reference values of anatomical and physiological data (ICRP, 2002). This provides a series of reference values for both male and female subjects of six different ages: newborn, 1 year, 5 years, 10 years, 15 years and adult. Anyone requiring this level of detail should consult *Publication 89*. Some examples of the reference data for an adult male are shown in Table 9.1.

It was pointed out above that the fate of a particular radionuclide inside the body depends on its chemical and physical form. For example, some elements distribute themselves fairly uniformly and so irradiate the whole body at about the same rate. The

Table 9.1 Some characteristics of reference man

(a) Organs of reference man

Organ	Mass (kg)	Percentage of total body
Total body	73	100
Skeleton	10.5	14
Muscle	29	40
Fat	14.6	20
Blood	5.6	7.7
Gastrointestinal tract (including contents)	2.3	3.2
Thyroid gland	0.02	0.027

(b) Water balance

Water intake (L/day)		Excretion (L/day)	
Foods and fluids	2.6	Urine	1.6
Oxidation	0.3	Sweat	0.5
		Insensible	0.69
		Faeces	0.11
Total	2.9	Total	2.9

(c) Air balance

Vital capacity of lungs	5.0 L
Air inhaled during 8-h working day	9.6 m^3
Air inhaled during 16 h not at work	13.3 m^3
Total	~23 m^3/day

These values are for the adult male. In most cases, the values for adult female are lower. For full details of the characteristics of reference man see ICRP Publication 89.

majority of elements, however, tend to concentrate in particular organs so that an intake of radioactivity may result in different dose rates to the various organs of the body. Examples of such elements are iodine, which concentrates in the thyroid gland, and plutonium, which concentrates in the lung or bone.

The dose rate to any organ is proportional to the amount of radioactivity in the organ and decreases as the radioactivity decays or is excreted. The decay of a radionuclide is exponential in character and it is found that the rate of excretion of most substances from the body may also be considered as approximately exponential. This means that an **effective decay constant** can be employed to describe the rate of removal of a radioactive substance from the body (see Fig. 9.1), namely:

$$\lambda_{eff} = \lambda_r + \lambda_b$$

where λ_r = the radioactive decay constant and λ_b = the biological decay constant.

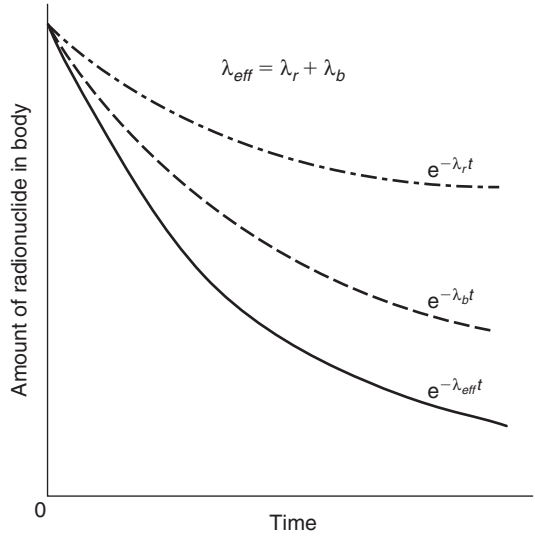


Figure 9.1 Typical elimination curve of a radionuclide in the body.

Since the decay constant is equal to $\log_e 2$/half-life, this equation becomes:

$$\frac{1}{T_{eff}} = \frac{1}{T_r} + \frac{1}{T_b}$$

where T_{eff} = effective half-life of a radioactive substance in the body, T_r = radioactive half-life of the substance and T_b = biological half-life of the substance.

Figure 9.2 illustrates the variation of dose rate with time following an intake of a radionuclide. The initial rise in the curve covers the period during which the nuclide is being transported to the organ of interest. At the peak, most of the radionuclide that is destined for the particular organ has reached it and the organ is receiving its maximum dose rate. Subsequently, the dose rate to the organ decreases approximately exponentially

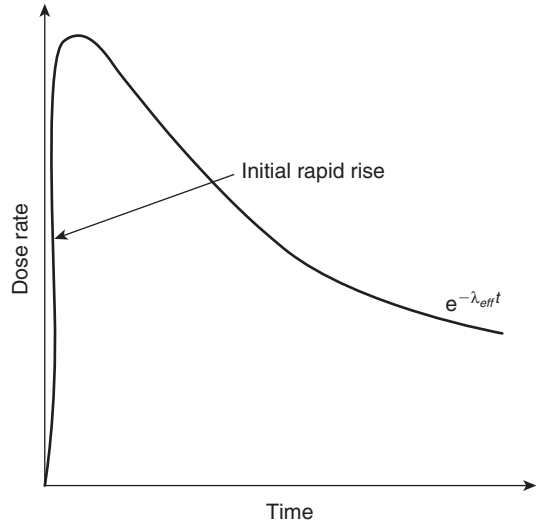


Figure 9.2 Variation of dose rate with time following an intake of a radionuclide.

as the radionuclide decays and is excreted. The total dose received by the organ is obtained by evaluating the area under the curve. Thus, a given intake of a radionuclide will 'commit' the organ (or organs) at risk from that nuclide to a certain dose, which is known as the **committed equivalent dose**. This depends on the initial dose rate and on the removal rate. It is usually assumed that a given intake of a particular radionuclide will result in the same committed equivalent dose whether it is received in a single large intake or a large number of small intakes. By applying the tissue weighting factor, the committed equivalent dose can be expressed as a committed effective dose.

For the majority of radionuclides, the dose is received over a relatively short period following an intake, typically a few months to a few years, because after this time the radioactivity will have decayed or have been excreted. For some long-lived species, such as plutonium isotopes, the radioactivity is excreted only very slowly and so the dose is received over a very long period. In this latter case, the committed equivalent dose is defined as that which is received in a period following the intake of 50 years for adults or 70 years for children, as these times represent the likely maximum lifespan of the individual following the intake.

9.3 DOSE PER UNIT INTAKE

Committee 2 of the ICRP, dealing with internal exposure, has calculated values of the committed effective dose for an intake of 1 Bq for virtually all radionuclides that could be of interest in radiation protection. These are referred to as **effective dose coefficients for intakes**. Separate values are given for intake by inhalation and ingestion. The basic values for adult workers are given in ICRP *Publication 68* (ICRP, 1994). Situations can occur in which it is necessary to assess the dose to members of the public, including infants and children, for example where they are exposed to low levels of radioactivity in the environment. The values for members of the public are tabulated in ICRP *Publication 72* (ICRP, 1995). It

should be noted that the dose coefficients given in *Publication 68* and *Publication 72* are based on earlier values of anatomical and physiological data, and also on earlier values of the tissue weighting factors. Revised values will be published by the ICRP in due course.

Table 9.2 illustrates some examples of the dose coefficients for some important radionuclides for inhalation and ingestion. Since the transfer of any material from the lung or from the gut is influenced by its chemical form, particularly its solubility, it is necessary to specify different values for different chemical forms of the majority of radionuclides. It can be seen that the committed effective dose from an intake of 1 Bq varies widely for different radionuclides reflecting the different types of emission (α or β), the half-life and the behaviour of the particular chemical form of a radionuclide in the body. Radionuclides with a high dose coefficient are of high radiotoxicity and those with a low value are low radiotoxicity emitters.

Table 9.2 Some values of effective dose coefficient for workers

Radionuclide	Compound	Effective dose coefficient (Sv/Bq)	
		Inhalation	Ingestion
Tritium	Tritiated water	1.8×10^{-11}	1.8×10^{-11}
	Organically bound	4.1×10^{-11}	4.2×10^{-11}
	Hydrogen gas	1.8×10^{-15}	–
Sodium-22	All	2.0×10^{-9}	3.2×10^{-9}
Iodine-131	All	1.1×10^{-8}	2.8×10^{-8}
Caesium-137	All	6.7×10^{-9}	1.3×10^{-8}
Plutonium-239	Oxides and hydroxides	3.2×10^{-5}	2.5×10^{-7}
	All other compounds	8.3×10^{-6}	9.0×10^{-9}

These values are for a particle size of 5 μm and are taken from ICRP Publication 68, (ICRP, 1994). *The corresponding data for members of the public are given in ICRP* Publication 72, (ICRP, 1995).

It should be noted that the values recommended by the ICRP are revised periodically in the light of new scientific data and so values from current sources should always be used for radiation protection purposes.

Example 9.1

During a particular year, it is estimated that a worker has been exposed to intakes of 1.5×10^5 Bq of sodium-22 (via ingestion) and 50 Bq of plutonium-239 oxide (via inhalation). What is the total committed effective dose from the intakes?

For sodium-22, the ingestion dose coefficient (see Table 9.2) is 3.2×10^{-9} Sv/Bq and for plutonium-239 in oxide form the inhalation dose coefficient is 3.2×10^{-5} Sv/Bq. The committed effective dose from the intakes is then:

For ^{22}Na: 1.5×10^5 Bq $\times$ 3.2×10^{-9} Sv/Bq = 4.8×10^{-4} Sv = 0.48 mSv

For ^{239}Pu: 5.0×10^1 Bq $\times$ 3.2×10^{-5} Sv/Bq = 1.6×10^{-3} Sv = 1.6 mSv

The total dose is therefore 2.08 mSv.

It should again be emphasized that in radiation protection the primary requirement is not just to maintain doses within the dose limits but to ensure that doses are as low as reasonably achievable within those dose limits. This is particularly important in the case of internal radiation because of the greater difficulty in controlling exposure and of assessing the doses to individuals from intakes of radioactivity.

In assessing the total dose received by a person in a year, both the external and internal doses must be considered to ensure that the recommended dose limit is not exceeded. For this purpose, it is recommended that the committed dose be assigned to the year in which the intake occurred even though some, or perhaps most, of the dose will not be received for many years.

9.4 CONTROL OF THE CONTAMINATION HAZARD

9.4.1 Basic principles

As with external radiation, the consideration in the control of the radioactive contamination hazard is to ensure that doses are as low as reasonably achievable and that the relevant dose limits are not exceeded. However, the basic approaches to controlling exposure are quite different. In the case of external radiation, the dose rate in a working area can be easily measured and the dose received by workers can be continuously monitored and controlled using personal dosimeters. Where there is significant radioactive contamination, however, there is much greater uncertainty both in the levels of radioactivity on surfaces and in the air in the workplace and, particularly, in the quantities likely to be inhaled or ingested by a worker. The approach must therefore be to avoid the contamination of working areas wherever possible and clean up any releases that do occur. In many radioactive facilities, nevertheless, there will be situations where some exposure to contamination is unavoidable, for example when it is necessary to break into contaminated equipment for repairs or maintenance. In such situations, the approach is to protect the worker by means of appropriate clothing and respiratory protection.

The following hierarchy should be applied in controlling the radioactive contamination hazard:

1. Eliminate the use of the radioactive material if possible.
2. Minimize as far as possible the amount of activity being handled.
3. Contain the radioactive material: normally at least two levels of containment are provided.
4. Follow the correct procedures, including the use of washing and monitoring facilities.
5. Use appropriate personal protective clothing and equipment.
6. Clean up contamination as soon as it occurs.

Figure 9.3 illustrates a typical containment system which might be applied in the relatively simple case of a radiochemistry laboratory. The four levels of containment are the bottle containing the liquid, the splash tray, the fume hood and, finally, the barrier at the entrance to the laboratory.

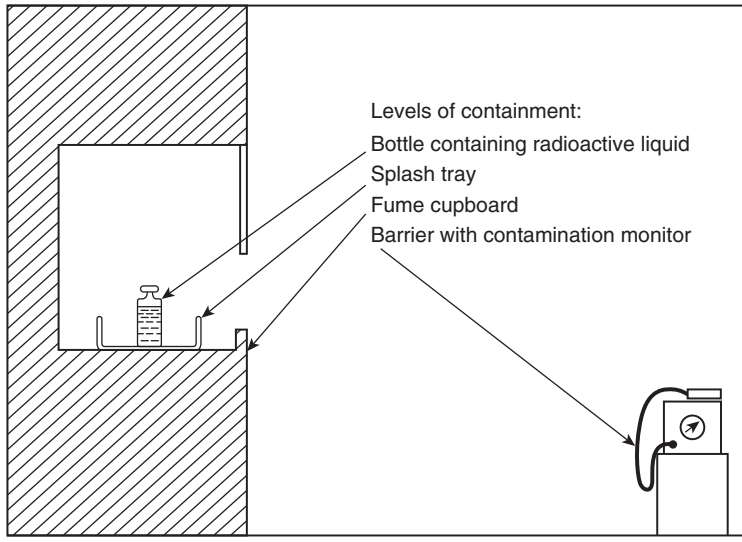


Figure 9.3 Schematic diagram illustrating four levels of containment.

9.4.2 Area classification

As with the external radiation hazard, routine control of contamination is by means of a system of **area classification**. Table 9.3 shows the basis on which areas should be classified. As can be seen, **supervised areas** are those in which contamination is not normally expected but which could occur as a result of some failure in equipment or procedures. They provide a useful buffer zone between **controlled areas**, in which contamination is likely to be present to a greater or lesser extent, and **uncontrolled areas**. Within a controlled area, there could be areas where the contamination hazard is very high and where additional controls are imposed. The essential point is that the system should be designed to provide a safe but practical system of working in the particular conditions.

Table 9.3 Control levels for area classification

Type of area	Radiological conditions
Uncontrolled (non-active)	No potential for radioactive contamination
Supervised (contamination)	Low potential for contamination but need to keep under review
Controlled (contamination)	Contaminated to greater or lesser extent and requiring appropriate precautions and protection measures

Whenever possible, contamination should be cleaned up as soon as it occurs. This prevents further spread that makes the eventual decontamination more difficult.

Regular surveys should be made in supervised and controlled areas and in the adjacent uncontrolled areas to ensure that contamination is not spreading beyond the barriers.

9.4.3 Protective clothing

The protective clothing requirements in a contaminated area depend on the nature and amount of the contamination. In supervised areas, where there is a low potential for

surface contamination, an ordinary laboratory coat with overshoes and gloves will often be sufficient. In controlled areas, the standard of protective clothing and equipment will be entirely dependent on the conditions. Where there is the potential for airborne contamination, some form of respiratory protection, such as a filter mask or a mask with an air supply, would normally be worn. Where the radioactive contamination is in the form of a contaminated liquid, the worker would normally be required to wear a fully enclosed PVC suit with a filter mask or fresh air supply. Some examples of protective clothing are illustrated in Figure 9.4.

Figure 9.4 Protective clothing and equipment. (a) Coverall. (b) Coverall and gloves. (c) Hooded coverall, gloves and respirator. (d) Pressurized suit (courtesy of NDA photo library, reproduced with permission).

Whatever the standard of protective clothing, the change area and barrier arrangements must be efficient and should have the following facilities:

1. hand-wash basin (and possibly a shower) and monitoring instruments (for example, a hand and clothing monitor);
2. suitable stowage on the non-active side of the barrier for the worker's personal clothing;
3. conveniently placed protective clothing ready for use;

4. containers for used clothing and radioactive waste;
5. noticeboards at the barrier stating 'no unauthorized entry', the hazards in the area, the clothing to be worn and any other precautions to be taken;
6. emergency instructions posted in the area, detailing actions in the event of possible incidents such as criticality, fire or serious personal contamination (consideration must also be given to the provision of suitable emergency exits).

Special arrangements have to be made for laundering clothing worn in contaminated areas and the effluent from laundry facilities is treated as liquid radioactive waste.

9.4.4 House rules and training of personnel

The control of contamination depends on everyone who enters a controlled or restricted area and so all personnel who work in such areas should be given initial and subsequent periodic training in the hazards involved and in the house rules. Some typical house rules for controlled and restricted areas are as follows.

1. No eating, drinking or smoking.
2. No mouth operations (such as pipetting).
3. Any wounds should be covered with a waterproof dressing before entering the active area. This is most important since open wounds provide a direct route for contamination into the bloodstream.
4. Wounds sustained in the area should be reported to the person in charge and treated immediately.
5. Ordinary handkerchiefs should not be used in an active area. Disposable tissues should always be available.
6. All items being removed from an active area should be subject to health physics clearance before being permitted to leave the area. Active items should be suitably labelled. Whenever possible, sets of tools and cleaning gear should be reserved solely for use in active areas and should be clearly marked as 'active'.

9.5 RADIOTOXICITY AND LABORATORY CLASSIFICATIONS

A toxicity classification for radionuclides was recommended by the International Atomic Energy Agency (*IAEA Technical Reports Series No. 15*) to act as a guide to the procedures and facilities required for handling unsealed radioactive substances. In the classification scheme, radionuclides are divided into four groups according to their radiotoxicity:

Group I – high (e.g. ^{239}Pu and ^{241}Am)

Group II – upper medium (e.g. ^{90}Sr and ^{131}I)

Group III – medium and lower medium (e.g. ^{32}P and ^{65}Zn)

Group IV – low (e.g. ^{129}I, natural uranium)

Note that there are many nuclides in each group and the nuclides mentioned above serve as common examples.

In addition, three classes of laboratory, classes 1, 2 and 3, are defined by the IAEA. A class 1 laboratory is a specially designed facility with elaborate equipment to enable the safe handling of high levels of radioactivity. Class 2 laboratories are of a standard comparable

to any high-quality chemistry laboratory, whereas class 3 includes ordinary laboratories not originally designed for handling toxic materials. The quantities of radionuclides that can be handled with reasonable safety in the various classes of laboratory are shown in Table 9.4.

Table 9.4 Guide to quantities of radionuclides that may be handled

Radionuclide toxicity	Class of laboratory		
	3	2	1
Group I	Up to 370 kBq	Up to 370 MBq	Above 370 MBq
Group II	Up to 3.7 MBq	Up to 3.7 GBq	Above 3.7 GBq
Group III	Up to 37 MBq	Up to 37 GBq	Above 37 GBq
Group IV	Up to 370 MBq	Up to 370 GBq	Above 370 GBq

These figures are for normal, wet chemical operations. Modifying factors may be applied as follows:

Work	Modifying factor
Storage in closed, vented containers	× 100
Simple wet chemistry; low specific activity	× 10
Complex wet or simple dry operations	× 0.1
Dry and dusty operations	× 0.01

9.6 DESIGN OF AREAS FOR RADIOACTIVE WORK

A considerable amount of pre-planning must be done before an active laboratory or active area is set up. Apart from the general features of design expected in any good facility, special attention must be paid to ventilation and to surface finishes. Within the facility air should flow from the lowest activity areas to the highest activity areas. Ventilation systems require efficient filtration units to remove particulate activity before discharge. In the case of gaseous activity, which would not be removed by filtration, great care must be exercised in the location of outlets to ensure adequate dispersal of any discharged activity.

Surfaces in an active area should be smooth and unbroken and be made from materials that are chemically inert, non-absorbent and water repellent. Consideration must also be given to possible decontamination problems that might arise, and so materials must be chosen which are either easily decontaminated or which can be conveniently removed and replaced.

9.6.1 Walls, floors and ceilings

The basic requirement is that the walls, floors and ceiling should have a good clean finish and be free from gaps or cracks in which contamination could accumulate. From the point of view of cleanliness and ease of cleaning, it is desirable to have covings at all angles of walls, ceiling and walls, and floors and walls.

Plastered walls and ceilings can be made non-porous and smooth by the application of gloss paint. An alternative approach is to ‘face’ the walls with a suitable strippable material, for example melamine laminate on a resin-bonded plywood backing which

is butt jointed with the joints sealed. Other finishes that have been used for walls and ceilings include chlorinated rubber-based paint, epoxide resin paint and various other strippable materials.

The most satisfactory floor covering consists of sheet PVC which is stuck down with all joints welded. Such a floor covering should have an integral coved skirting. An alternative covering consists of sheet linoleum which is made water repellent by applying a hard wax followed by a soluble wax. In the case of sheet linoleum, a separate pre-formed coved skirting has to be used and the joints have to be cold-welded. PVC or linoleum tiles are not normally recommended because the many joints in the floor make cleaning up after contamination a difficult operation. Concrete and wood are poor floor materials but their use is sometimes unavoidable. When they have to be used, they should be treated with a rubber-based paint to make them water repellent.

9.6.2 Working surfaces

Working surfaces should be finished in hard non-porous materials which have the necessary heat- and chemical-resisting properties. The most commonly used materials include the following.

1. Melamine resin plastic laminate such as Formica. It should be bonded to the backing material with a resin glue to give the necessary temperature resistance.
2. PVC sheet, such as Darvic, which can be welded and is completely self-extinguishing.
3. Stainless steel is a useful material but there is a tendency to get physical bonding between it and corrosion products. Also, stainless steels are susceptible to attack from certain chemicals, for example hydrochloric acid.
4. Glass fibre-reinforced resin which can be moulded to shape. It can be treated to make it fire resistant, but it may burn in a fire.
5. Polypropylene, which can be welded and heat-formed. This material has such a high chemical resistance that it is difficult to find suitable adhesives for it. It is not fire resistant and, once ignited, will continue to burn.

9.6.3 Glove boxes

These are installed in active laboratories to facilitate the handling of hazardous materials. They consist of a leak-tight enclosure in which objects or materials may be manipulated through gauntlet gloves attached to ports in the walls of the box (see Fig. 9.5). Their aim is to provide containment for materials that are either radioactive or chemically toxic, or both. Usually, they do not provide any shielding protection against penetrating radiation and so they are used for α or β emitters. When γ-emitting isotopes must be handled, a wall of lead bricks is usually constructed between the operator and the glove box. In some cases, windows containing zinc bromide are also used to provide shielding.

Glove boxes are maintained at a pressure slightly below that of the outside laboratory. This means that air will flow into the glove box should a leak develop, thus preventing the contamination from escaping. Two filters are normally placed in the ventilation system, one to remove dust from the air being drawn into the glove box and the second to remove radioactive particles from the air being drawn out of the box.

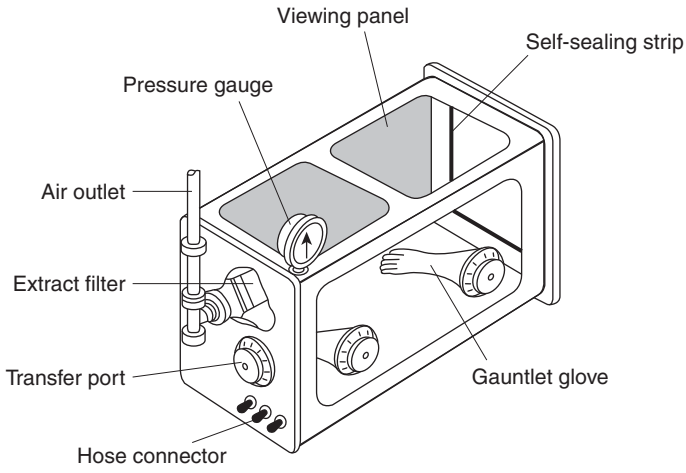


Figure 9.5 Schematic drawing of a glove box.

9.6.4 Fume cupboards

A fume cupboard is used when relatively low levels of activity (in the megabecquerel range) are being handled. The material is handled via an opening at the front through which air is drawn from the laboratory into the fume cupboard (see Fig. 9.6). This protects the operator from any leakage of contamination through the opening into the general

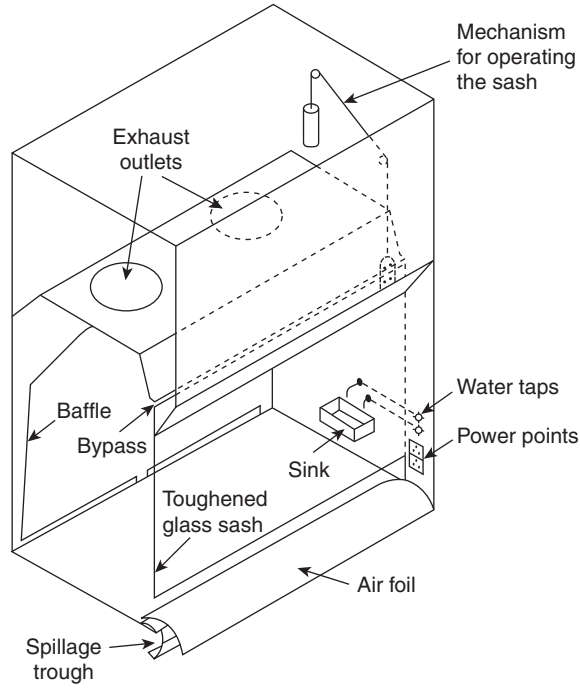


Figure 9.6 Schematic drawing of a fume cupboard.

laboratory. The services usually required in a fume cupboard are water, gas, vacuum and electricity. The controls for these services should be situated outside the fume cupboard to minimize the number of movements through the front opening. It is good practice to ensure that the openings of the fronts of fume cupboards are kept to a minimum to reduce the chance of radioactive contamination entering the general laboratory atmosphere.

9.7 TREATMENT OF CONTAMINATED PERSONNEL

Once a radioisotope has become lodged in the body, very little can be done to increase the rate of elimination (i.e. decrease the biological half-life) of the isotope. This means that every effort must be made to prevent contamination entering the body. To this end it is vital that all personnel should obey the house rules and always wear the correct protective clothing. Even so, contamination incidents are bound to occur and so a knowledge of the correct treatment is vital.

The first action when dealing with a contaminated person is to ascertain whether or not they are injured. If there is serious injury, then first-aid treatment must be given as quickly as possible. Following any necessary medical treatment, the next actions are aimed at removing the contamination before it becomes absorbed and lodged in the body. However, before decontamination can be started, a careful survey must be carried out over the entire body with a suitable contamination monitor to determine the location of the contamination. In the case of partial contamination, it is necessary only to decontaminate the affected areas. For example, if a person has received contamination on the hands or face, these areas would be wiped with wet wipes and washed with liquid soap and water before being monitored again for residual contamination. Should contamination still be present, the process is repeated several times, taking care to ensure that the skin is not damaged. If contamination is still present after repeated washing, other decontamination techniques could be attempted, for example creams could be applied or the area might be covered to encourage the body to sweat out the contamination.

In the case of whole-body contamination, the first action following the removal of protective clothing would normally be to wash the person's hair over a hand-wash basin. This removes contamination from the hair and prevents it being washed down into the mouth while undergoing whole-body decontamination under a shower using soap. After the shower, the person is monitored carefully and may need further showers before decontamination is complete. In addition, when the person has been exposed to excessive airborne contamination, other measures, such as nose-blowing and rinsing, may be required to remove the contamination completely.

If a minor wound is sustained in a contaminated area, it should be allowed to bleed freely and be copiously washed with water to encourage the removal of the contamination. If the person cannot be immediately and completely decontaminated, or if the wound is serious, medical assistance should be obtained as quickly as possible.

Following an intake by ingestion, substances designed to prevent or reduce absorption from the gastrointestinal tract, for example antacids or ion-exchange resins, may be administered promptly after the intake. If radionuclides of high toxicity, such as ^{239}Pu, are absorbed through a wound or inhaled in a soluble form, certain chemicals called chelating agents may be administered to promote excretion. Unfortunately, these substances tend to be chemically toxic themselves. If significant quantities of nuclides of high toxicity, such as

^{239}Pu, are inhaled in an insoluble form, a procedure known as lung lavage may be carried out under medical supervision.

The absorption of certain radioisotopes can be blocked by the prior ingestion of substantial amounts of a stable isotope of the same element. For example, the uptake of radioiodine to the thyroid can be greatly reduced by previous ingestion of potassium iodate. The usual adult dose is 170 mg in tablet form. This has an important application in the event of a reactor accident.

9.8 CONTAMINATION MONITORING

9.8.1 Sensitivity

It has already been mentioned that quite small quantities of radioactivity which represent an insignificant external hazard can give rise to a significant internal hazard. This means that the radiation levels generated by an amount of radioactive contamination sufficient to cause an internal hazard are generally much lower than the levels that would cause an external radiation hazard. As a result, contamination monitors in general need to be more sensitive than radiation survey meters.

To fulfil this requirement for increased sensitivity, contamination meters are constructed from detectors which have their own built-in amplification system (Geiger–Müller or scintillation counters). The activity level is recorded as a counting rate (counts per second or per minute), and the response of the monitor to the relevant radionuclides must be known before the contamination level can be calculated (for example in becquerels per square centimetre).

9.8.2 Direct surface contamination monitoring

This is the simplest and most convenient method of contamination monitoring and is carried out to establish the presence of contamination on such surfaces as bench tops, clothing, skin and so on. Direct measurements allow the contamination level to be calculated in becquerels per square centimetre, or related to the derived limits of surface contamination. A typical contamination monitor consists of either a mains- or battery-operated ratemeter to which various types of detecting head can be connected.

Alpha contamination is detected by means of a zinc sulphide scintillator coupled to a photomultiplier tube. The zinc sulphide screen is covered with Melinex (DuPont), a very thin plastic material, coated with aluminium to make it light-tight. This covering has to be thin enough to allow α particles to penetrate through to the zinc sulphide screen. It is important when carrying out direct surface monitoring for α contamination to keep the probe as close to the surface as possible to record the true activity level.

Beta-contamination monitors usually use plastic phosphor scintillators as the detecting medium. These may be used in conjunction with a zinc sulphide screen to form a dual probe which permits simultaneous α and β monitoring (Fig. 9.7). A single photomultiplier tube is used with a circuit to differentiate between the α and β pulses. Another type of β detector uses a Geiger–Müller tube in a suitable holder or probe. The probe has a shutter which should be opened for monitoring. It is difficult to detect low-energy β contamination, for example from ^{3}H, ^{14}C and ^{63}Ni, and care must be taken when selecting an instrument for this purpose. Alternatively, monitoring for low-energy β-emitters may be by means of smear samples (see below).

Figure 9.7 Dual (α and β) contamination probe and ratemeter.

Beta probes respond to γ radiation, which makes direct contamination monitoring difficult in areas of high γ background. Under such circumstances indirect methods have to be used.

9.8.3 Smear surveys

Smear surveys are an indirect method of measuring surface contamination levels. They are used to detect very low levels of contamination, to monitor for contamination in an area of high radiation background or to monitor for radionuclides that are difficult to detect using direct monitoring methods. A filter paper is wiped over a known surface area, usually 0.1 m^2 (about 30 cm × 30 cm), placed in a polythene envelope to avoid cross-contamination and then taken to an area of low-radiation background for assessment (see section 10.2.4). Here it is counted in a detecting system of known efficiency. The contamination level can then be calculated from the formula:

$$\text{Contamination level (Bq/cm}^2) = C_c \times \frac{100}{E_c} \times \frac{1}{A} \times \frac{100}{E_F}$$

where C_c = count rate, corrected for background, in counts per second, E_c = overall percentage efficiency of the counting system, A = area smeared in cm^2 and E_F = percentage of the contamination picked up by the paper.

The last quantity, E_F, is quite difficult to determine and is not very reproducible. It is dependent on various parameters, such as the physical and chemical nature of the contamination, the nature of the base surface and so on. In some circumstances, E_F is

taken as 100 per cent and in these cases it is the 'removable' contamination which is being determined. More usually, a figure of 10 per cent is assumed.

A useful qualitative technique commonly used within active areas is to swab a large surface area using a damp paper towel and then to monitor the swab. This technique has the advantage that it also decontaminates the surface to some degree.

9.8.4 Air monitoring

Air monitoring is carried out in areas where airborne contamination may occur. There are basically three ways by which contamination can become airborne:

1. disturbance of surface contamination on the surfaces in the active area;
2. allowing liquid contamination to dry out;
3. carrying out dry, dusty operations, such as cutting, which cause the release of particulate activity.

Particulate airborne activity is measured by drawing a known volume of air through a filter paper (Fig 9.8). The filter paper is then counted in a low background area in precisely the same way as for smear survey papers (see section 10.2.4). The level of airborne activity is calculated from the count rate on the filter paper by means of the formula:

$$\text{Airborne contamination level (Bq/m}^3\text{)} = C_c \times \frac{100}{E_c} \times \frac{1}{V}$$

where C_c = corrected count rate in counts per second, E_c = overall percentage efficiency of the counting system and V = volume of air sampled in m^3.

Gaseous activity is normally measured by drawing a certain volume of air through a filter paper into a sample chamber, which is then sealed. The filter paper removes particulate activity so that the activity in the sample chamber is caused by radioactive gases only. The chamber is counted in a low background area and the gaseous activity level can be calculated.

Figure 9.8 Particulate air sampler (courtesy of Bird and Tole Air Sampling Products).

9.9 PERSONAL MONITORING

In most circumstances, airborne and surface contamination monitoring in the general work area is sufficient to ensure that intakes of contamination by personnel working in an area are insignificant. However, in some circumstances, such as working at glove boxes, general area air monitoring could seriously underestimate the concentration to which an operator is exposed and, in such cases, the operator would be equipped with a portable personal air sampler that draws air from close to his or her face. There are also situations when it is necessary to measure directly any radioactivity that has been inhaled or ingested. Examples of such circumstances are:

1. The annual limit of intake of the radioisotope in question is very low (for example, plutonium).
2. The isotope is difficult to detect by normal methods.
3. Individuals are working in areas where there is a significant risk of them receiving an intake.
4. An accident has occurred.

The type of monitoring used will depend on the type of radioisotope in the body.

Gamma-emitting isotopes can be measured in a whole-body counter, where the subject is placed in a low-background, shielded facility and the γ emission is detected by several large-volume sodium iodide (NaI) scintillation counters or HPGe detectors.

Alpha- or β-emitting isotopes are measured by excretion monitoring, for example faeces (ingestion of insoluble contamination) and urine (soluble contamination).

Following a suspected inhalation of activity, monitoring of nasal swabs or nose-blows is a useful indicator of whether or not a significant intake has occurred.

SUMMARY OF KEY POINTS

Internal radiation hazard: caused by radioactive materials inside the body.

Routes of entry: inhalation, ingestion and direct entry through wounds in the skin.

Effective decay constant: $\lambda_{eff} = \lambda_r + \lambda_b$.

Committed equivalent dose to an organ: the equivalent dose to which an organ is committed following an intake.

Committed effective dose: the committed equivalent dose to an organ multiplied by the appropriate tissue weighting factor.

Effective dose coefficient: the effective dose from an intake of 1 Bq of a radionuclide. ICRP has tabulated values for all radionuclides of interest for both inhalation and ingestion.

Surface contamination monitoring: direct monitoring, zinc sulphide for α detection, plastic phosphor scintillating or Geiger–Müller tubes for β detection. Indirect monitoring using smear surveys.

Airborne contamination monitoring:

1. **Particulate** contamination level determined by drawing a known volume of air through a filter paper.

2. **Gaseous** activity measured by drawing a known volume of the atmosphere through a filter paper into a sample chamber and counting the activity.

Personal monitoring:

1. γ emitters, whole-body counter;
2. α or β emitters, faeces, urine or breath monitoring; and
3. personal air sampling.

REVISION QUESTIONS

1. If a worker is exposed to intakes of 2×10^5 Bq of iodine-131 (via inhalation) and 10^6 Bq of caesium-137 (via inhalation) in 1 year, what is the maximum external dose that he can receive in the year within a 20-mSv annual dose limit?
2. Calculate the maximum intake of plutonium-239 oxide which a worker may receive via inhalation in a year if his dose from external exposure is 10 mSv.
3. Discuss the problems encountered in direct surface contamination monitoring.
4. Calculate the surface contamination level from the following data:

Uncorrected count rate on smear paper	3840 counts/min
Background count rate	240 counts/min
Efficiency of counting system	15 per cent
Area of surface smeared	0.1 m^2
Pick-up efficiency of smear	10 per cent

5. Calculate the level of airborne particulate activity from the following data:

Uncorrected count rate on filter paper	29 832 counts/4 min
Background count rate	1032 counts/4 min
Efficiency of counting system	12 per cent
Flow rate through particulate sampler	0.03 m^3/min
Time for which sampler was run	2 min

6. Discuss the issues involved in dealing with a contaminated person who is also injured.
7. What is meant by the 'radiotoxicity' of an isotope? Discuss how this affects the classification of laboratories that handle radioactive material.

10 Practical health physics techniques

10.1 BASIC TECHNIQUES

In this chapter, some of the basic techniques required in health physics laboratories for the quantitative and qualitative analysis of samples are discussed. These include gross alpha (α) and beta (β) counting of samples and the use of spectrometry and other methods for identifying radionuclides. Methods for leak testing radioactive-sealed sources are also described.

10.2 ANALYSIS TECHNIQUES

10.2.1 Identification of unknown samples

In general, the assessment of a radioactive sample, such as an air sample filter paper, requires not only measurement of the sample activity but also identification of the radionuclides present. Indeed, in most cases precise measurement of the sample activity is not possible without some knowledge of the nuclide and hence the type and energy of radiation emitted. The sample must, therefore, be analysed or else some assumptions must be made about its composition. It is not usually practicable or necessary to analyse every sample taken and so, in a particular area, occasional samples are analysed and the results are applied to other samples taken under similar conditions, perhaps with the introduction of a safety factor.

The characteristics of a nuclide that might possibly be determined in order to enable its identification are the type and energy of radiation emitted and the half-life. The methods by which these measurements may be made are described in the following sections.

10.2.2 Energy determination

The most convenient method of energy determination is **gamma (γ) spectrometry**, which was previously described in section 7.6.4. For many commonly encountered radionuclides, the observed spectrum enables rapid identification and measurement of sample activity. Similar techniques may be used for α- and β-emitters but the equipment is less readily available.

If the nuclide is a pure β-emitter or if the equipment available does not include a γ spectrometer, reliance is often placed on **β-absorption methods**. These involve counting a sample in a β-counting system (e.g. a Geiger–Müller detector in a shielded castle) and observing the count rate with various thicknesses of aluminium placed between the sample and the detector. An absorption graph is obtained by plotting the count rate against the absorber thickness, usually on log-linear graph paper (Fig. 10.1).

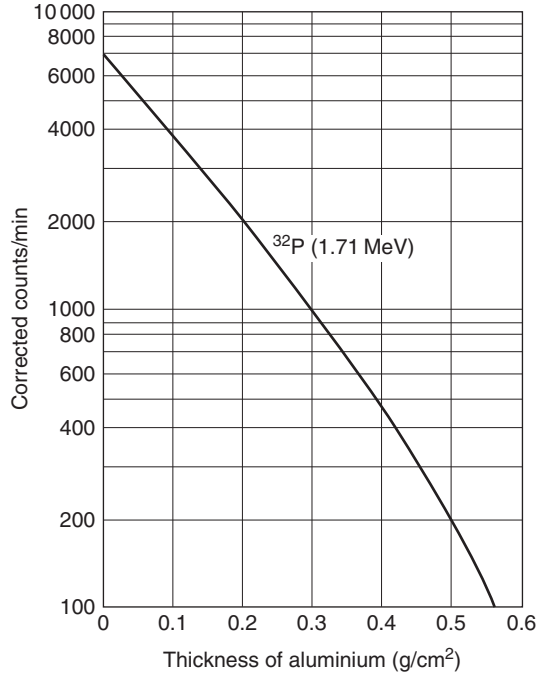


Figure 10.1 Typical β-absorption curve.

The basis of this method is that low-energy β radiation is more easily absorbed than high-energy radiation and so, by comparison of the absorption curve with curves for β radiation of a known energy, the β energy of the sample may be estimated.

A simpler method is to take a series of counts with increasing absorber thickness until the count rate has been reduced to about one-quarter of the initial value. The corrected count rate (see sections 10.2.4 and 10.2.5) is then plotted against absorber thickness and the thickness that would reduce the count rate to one-half of the initial value is read off the bottom scale. This is the **half-value thickness** (**HVT**) for that β energy. An example of the method is illustrated in Figure 10.2. The thickness of aluminium is expressed in units of grams per square centimetre (g/cm^2). This is obtained by multiplying the thickness of aluminium (cm) by the density (g/cm^3) to obtain g/cm^2.

The relationship between the β-particle maximum energy and HVT is shown in Figure 10.3. Considering the sample in Figure 10.2, the HVT is 0.074 g/cm^2 and so, from Figure 10.3, the β energy is found to be about 1.4 MeV.

It should be mentioned that such determinations are not always easy because, in practice, samples often contain more than one nuclide, and consequently several β-ray energies may be present. Even assuming that the β energy is determined, this is probably insufficient to identify the radionuclide positively. A further indication may be obtained, in some cases, by measuring the half-life of the nuclide.

10.2.3 Determination of half-life

The concept of half-life of a radionuclide was introduced in section 2.7. It will be recalled that the activity of a sample at time *t* is given by:

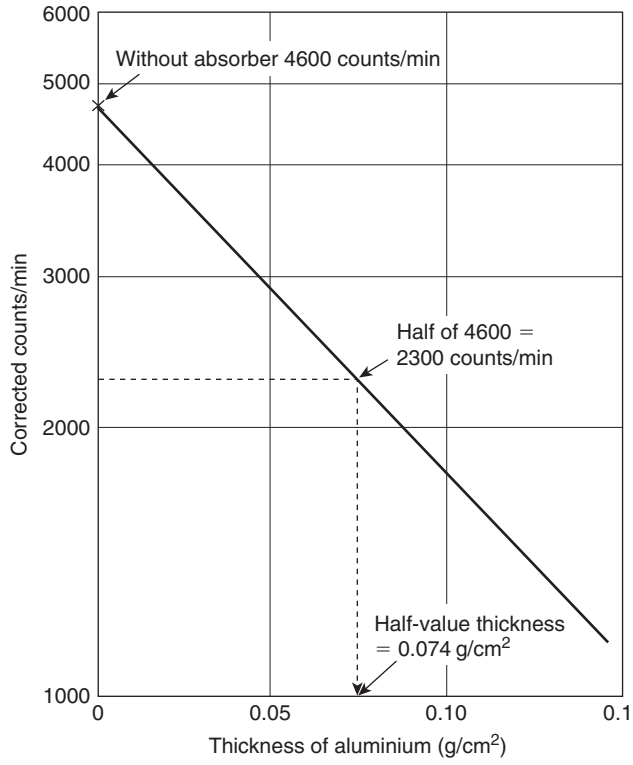


Figure 10.2 Determination of half-value thickness for a β emitter.

$$A_t = A_0 e^{-\lambda t}$$

where A_0 = the initial activity and λ = radioactive decay constant for the nuclide.

The half-life $T_{½}$ is the time at which A_t is one-half of the value of A_0, thus:

$$\frac{A_t}{A_0} = 0.5 = e^{-\lambda T_{½}}$$

$$\therefore \log_e 0.5 = -\lambda T_{½}$$

$$-0.693 = -\lambda T_{½}$$

$$\therefore T_{½} = \frac{0.693}{\lambda} \text{ or } \lambda = \frac{0.693}{T_{½}}$$

It will be seen that the radioactive decay law can be written in an alternative form:

$$A_t = A_0 \, e^{-0.693t/T_{½}}$$

In cases of nuclides having half-lives of between a few minutes and a few months, the half-life can be determined by taking a series of counts on the sample at suitable intervals

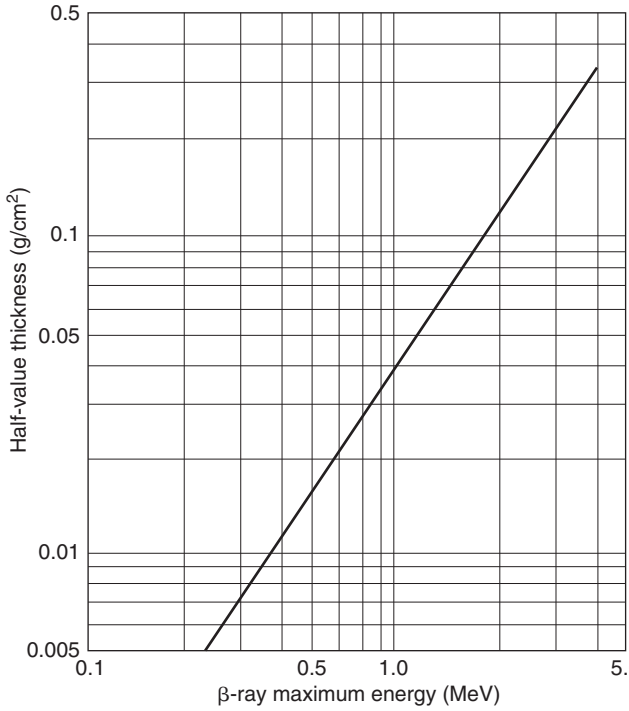


Figure 10.3 Relationship between half-value thickness and β-ray maximum energy.

(such that the count rate decreases by 10–15 per cent between counts). The corrected count rate is then plotted against time on log-linear graph paper, giving a straight line. The time to reduce the count rate to one-half of the initial value can then be read from the graph.

Example 10.1

Using log-linear graph paper, plot a decay curve from the following data and estimate the half-life of the nuclide:

Time (h)	0	2	4	8	12	18
Count rate (counts/min)	6720	6050	5690	4563	3930	2989

It will be found from this plot that the count rate decreases to one-half of the initial value (i.e. to 3360 counts/min) in about 15.4 h. The sample used to obtain these data is the same sample as used to obtain the absorption curve in Figure 10.2, and so it is known that the nuclide emits β particles of a maximum energy of about 1.4 MeV and decays with a half-life of 15.4 h. It should now be possible to identify the nuclide by referring to tabulations of nuclide data in order of half-life or β-particle energy. The nuclide present in the sample was, in fact, sodium-24, which has a maximum β-particle energy of 1.39 MeV and a half-life of 15.4 h.

As with β-absorption measurements, the procedure is often more difficult in practice because of the presence of more than one nuclide. This is illustrated in Figure 10.4, which

again shows a decay plot for sodium-24 although in this case a short-lived nuclide is also present. After an initial relatively rapid decay, the rate of decay decreases and eventually gives a straight line. The contribution from the longer-lived nuclide to the initial count rate can be obtained by extrapolating the straight line back to zero time. The half-life of this nuclide can then be obtained as before. To determine the half-life of the short-lived component, it is necessary to subtract the extrapolated values of the count rate due to the long-lived nuclide from the observed count rate (i.e. the difference between the curve and the straight line is plotted). This procedure gives the decay plot for the short-lived nuclide and the half-life can then be determined. In Figure 10.4, the short-lived nuclide is chlorine-38, which has a half-life of 37.3 min.

10.2.4 Gross alpha and beta counting

In general, it is necessary to use a high-sensitivity counting system comprising a radiation detector, a stabilized power supply, an amplifier, a discriminator and a scaler which registers the pulses received. Samples can be counted using an α detector (usually in the form of a drawer assembly) or a β detector (either as a drawer assembly or in a β castle) coupled with a scaler counter. The count rate can be converted to an activity using the formula:

$$\text{Activity (Bq)} = C_c \times \frac{100}{E_c}$$

where C_c = count rate, corrected for background, in counts per second, and E_c = overall percentage efficiency of the counting system.

The fraction of particles counted compared with the total number emitted is known as the **efficiency of the counting system**. The efficiency of a counting system is usually

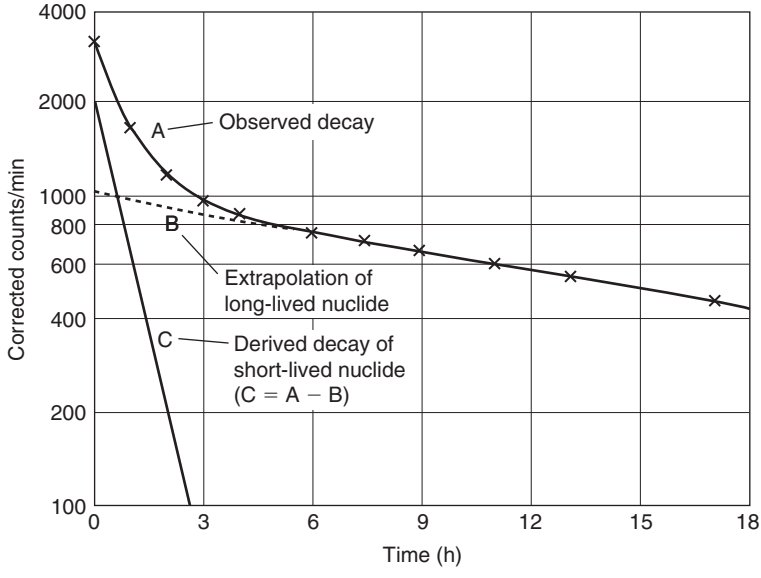


Figure 10.4 Decay curve of mixed ^{38}Cl and ^{24}Na sample.

determined by placing a standard source in the appropriate counting position and determining the number of counts recorded in a fixed time. This is then divided by the emission rate of the source to obtain the counting efficiency. Usually, the efficiency is multiplied by 100 and expressed as a percentage.

Example 10.2

A standard source of 220 Bq gives an uncorrected count of 2045 counts/min in a Geiger–Müller castle in which the background count is 65 counts/min. What is the efficiency of the system?

$$\text{Efficiency (\%)} = \frac{\text{corrected counts/min}}{\text{source emission rate, disintegrations/min}} \times 100$$

Corrected count rate = 2045 – 65 = 1980 counts/min. Now 220 Bq corresponds to 1.32×10^4 disintegrations per minute (dis/min). Therefore:

$$\text{Efficiency} = \frac{1980 \times 100}{1.32 \times 10^4} = 15\%$$

Example 10.3

Calculate the activity of a source which gives an uncorrected count rate of 4925 counts/min in the equipment mentioned in the previous example.

$$\text{Activity (Bq)} = \frac{\text{corrected count rate (cps)}}{\text{efficiency (\%)}} = 100$$

Corrected count rate = 4925 – 65 = 4860 counts/min = 81 counts/s. Therefore:

$$\text{Activity} = \frac{81}{15} \times 100 = 540 \text{ Bq}$$

In certain cases it is possible to mount a very thin radioactive sample at the centre of the sensitive volume of the detector and count all the particles or photons emitted; this is known as **4π geometry counting**. In the more common counting systems, only a fraction of the particles or photons emitted can enter the detector. Apart from the geometry of the counting system, a number of other factors influence the counting efficiency. These include backscatter, self-absorption in the source, absorption in the counter window, and absorption in the air gap between the source and the detector, all of which depend to some extent on the energy of the radiation. Ideally, the instrument should be calibrated with a source of the same nuclide and the same geometry as the samples to be counted. In the evaluation of routine samples for health physics purposes, a high degree of accuracy is not normally required and it is usual to calibrate the equipment with one typical source and use the same calibration for the majority of samples. It is important to be aware that some nuclides, particularly low-energy β emitters, can be seriously underestimated by this procedure.

The counts registered on the scaler must be corrected for background and may need to be corrected for the resolving time of the apparatus. It might also be necessary to calculate the appropriate statistical error.

Analysis of a smear sample allows the **contamination level** to be calculated from the formula:

$$\text{Contamination level (Bq/cm}^2) = C_c \times \frac{100}{E_c} \times \frac{1}{A} \times \frac{100}{E_F}$$

where C_c = count rate, corrected for background, in counts per second, E_c = overall percentage efficiency of the counting system, A = area smeared in cm^2 and E_F = percentage of the contamination picked up by the smear paper.

The last quantity, E_F, is quite difficult to determine and is not very reproducible. It is dependent on various parameters, such as the physical and chemical nature of the contamination, the nature of the base surface and so on. In some circumstances, E_F is taken as 100 per cent and in these cases, it is the 'removable' contamination which is being determined. More usually, a figure of 10 per cent is assumed.

By measuring the count rate on an air-sample filter paper, the **airborne activity concentration** can be calculated by means of the formula:

$$\text{Airborne contamination level (Bq/m}^3) = C_c \times \frac{100}{E_c} \times \frac{1}{V}$$

where C_c = corrected count rate in counts per second E_c = overall percentage efficiency of the counting system and V = volume of air sampled in m^3.

Care must be taken when interpreting this information as radon daughters may also be present on the filter paper and hence result in an enhanced count rate on the filter paper. Instruments specifically designed to compensate for radon daughters can be used in areas where it is known that radon gas is present. Alternatively, the sample can be left to decay and recounted once the radon daughters have decayed away (typically a few days).

10.2.5 Corrections for resolving time

When a charged particle or photon produces an interaction in the sensitive volume of a detector, there is a short period (usually of the order of 100 μs) during which further events cannot be recorded. This time is known as the **resolving time of the apparatus** and is the time required for the ions to be collected. It is sometimes referred to as the dead time of the detector because during this time it cannot respond to any new event. In many counting systems, it is easier to introduce a fixed dead time into the system than to determine the actual dead time experimentally. Since this artificial dead time is a function of known circuit parameters, it has a precisely defined value.

Consider a circuit which has a dead time of 200 μs and suppose that it is used to count a sample which gives an observed counting rate of 500 counts/s. In recording 500 counts in 1 s, the counter has been shut down for 200 μs after each count and was therefore inoperative for 500 × 200 μs (= 100 000 μs = 0.1 s). The 500 counts were therefore recorded in a counting time of only 0.9 s and so the true counting rate is:

$$\frac{500}{0.9} = 555 \text{ count/s}$$

This is written mathematically as:

$$C = \frac{c}{1 - ct}$$

where C = true count rate, c = observed count rate and t = dead time.

Care should be taken with the units in this equation; if C and c are in counts/s, t must be in seconds (e.g. 200 μs = 200×10^{-6} s).

It is inadvisable for dead time corrections of much greater than 10 per cent to be applied, so when the counting rate is very high it is better to reduce it by changing the geometry.

Example 10.4

Calculate the true counting rate of a sample if the observed counting rate is 15 000 counts/min and the dead time is 300 μs.

Therefore:

$$\text{True counting rate} = \frac{15\,000}{1 - 15\,000 \times 5 \times 10^{-6}} = 16\,216 \text{ counts/min}$$

10.2.6 Counting statistics

The radioactive decay of single atoms is random in time and so the number of particles or photons counted in a given time will fluctuate about an average value. The **standard deviation** (σ) is a measure of the scatter of a set of observations about their average value. In a counting system, if the average of a large number of counts is $\bar{N}$, then the standard deviation is found to be $\sqrt{\bar{N}}$. Thus, if:

$$\bar{N} = 400 \text{ counts}, \sigma = \sqrt{400} = 20$$

If a single count is made over a time t and N counts are recorded, the standard deviation on the count may be taken as $\sqrt{N}$. Usually, it is the counting rate which is of interest, and this may be written as:

$$\text{Counting rate} = \frac{N}{t} \pm \frac{\sqrt{N}}{t}$$

Example 10.5

A 5-s count on a sample gives a result of 100 counts. What is the counting rate and the standard deviation?

$$\text{Counting rate} = \frac{100}{5} \pm \frac{\sqrt{100}}{5}$$

$$= 20 \pm 2 \text{ counts/s}$$

The significance of the standard deviation is that 68 per cent of observations are within one standard deviation of the true counting rate. Hence, in the above example, there is a 68 per cent chance that the counting rate lies between 18 counts/s and 22 counts/s. The standard deviation is, therefore, a measure of the accuracy of an observation. In counting, greater accuracy can be achieved only by increasing the total count recorded. Thus, if in the previous example 1000 counts had been recorded in 50 s:

$$\text{Counting rate} = \frac{1000}{50} \pm \frac{\sqrt{1000}}{50}$$

$$= 20 \pm \frac{31.6}{50}$$

$$= 20 \pm 0.63 \text{ counts/s}$$

or if 10 000 counts were recorded in 500 s:

$$\text{Counting rate} = \frac{10\,000}{500} \pm \frac{\sqrt{10\,000}}{500}$$

$$= 20 \pm \frac{100}{500}$$

$$= 20 \pm 0.2 \text{ counts/s}$$

Although the counting rate is the same in all three cases, the accuracy has been improved by counting for longer periods.

The standard deviation can also be expressed as a percentage, for example on counts of 100, 1000 and 10 000:

$N = 100$	$\sigma = 10$ (10%)
$N = 1000$	$\sigma = 31.6$ (3.16%)
$N = 10\,000$	$\sigma = 100$ (1%)

For a count (or a count rate) to be accurate to a 1 per cent standard deviation, the counting period must be long enough to give at least 10 000 counts.

Usually, we need to find the counting rate S resulting from a source superimposed on a background counting rate. If N counts are recorded in time t_1 due to source and background, and B counts are recorded in time t_2 due to background alone, then the corrected counting rate S is given by:

$$S = \frac{N}{t_1} - \frac{B}{t_2} \pm \sigma_s$$

where:

$$\sigma_s = \sqrt{(\sigma_1^2 + \sigma_2^2)}$$

$$= \sqrt{\left(\frac{N}{t_1^2} + \frac{B}{t_2^2}\right)}$$

The general expression for this type of measurement becomes:

$$S = \frac{N}{t_1} - \frac{B}{t_2} \pm \sqrt{\left(\frac{N}{t_1^2} + \frac{B}{t_2^2}\right)}$$

Example 10.6

A source is counted in a Geiger–Müller castle and registers 6720 counts in 4 min. The background is then counted for 10 min and gives 480 counts. What is the corrected count rate and the standard deviation?

$$S = \frac{6720}{4} - \frac{480}{10} \pm \sqrt{\left(\frac{6720}{16} + \frac{480}{100}\right)}$$

$$= 1680 - 48 \pm \sqrt{(420 + 4.8)}$$

$$= 1632 \pm \sqrt{424.8}$$

$$= 1632 \pm 20.6 \text{ counts/min}$$

When counting very low activity sources, very long counting times may be required to achieve an acceptable statistical accuracy. Under such circumstances, it is desirable to choose the most efficient distribution of the time between the (source plus background) and the background count alone. The highest accuracy is achieved when:

$$\frac{t_1}{t_2} = \sqrt{k}$$

where t_1 is time spent counting (source plus background), t_2 is time spent counting background alone and k is the ratio of total counting rate of (source plus background) to the background rate alone.

It is often worthwhile, when confronted with a sample of unknown activity, to do a preliminary investigation of the counting rates to be expected. This entails a short (source plus background) count, followed by a background only count, each count being for 1 or 2 min duration only. From these results, the expected counting rates are determined roughly and the length and distribution of time for the accurate assessment of the sample are calculated.

Example 10.7

If the total counting rate for source plus background is 360 counts/min and the background counting rate is 40 counts/min, what proportion of the total available time should be spent on counting background?

$$\frac{\text{source + background count time}}{\text{background count time}} = \sqrt{\frac{\text{source + background count rate}}{\text{background count rate}}} = \sqrt{\frac{360}{40}} = 3$$

Therefore, one-quarter of the time should be spent on counting background.

10.3 LEAK TESTING OF SEALED SOURCES

Sealed sources should be tested at regular intervals (in the UK this is generally every 2 years) using an appropriate method to detect leakage of radioactivity from the source. Leak tests should normally be carried out by wiping the surface of the source directly with smear paper. The method used should be appropriate for the level of activity of the source and might, for example, require the use of suitable shielding and source handling tools, etc. The smear paper should then be assessed for the presence of contamination using an analysis technique suitable for the particular radionuclide.

There are some situations, however, when it may not be possible to carry out a direct leak test of a radioactive source, for example when the position of the sealed source within an article is such that it is not accessible or if the radiation dose to the person carrying out the leak test would not be as low as reasonably achievable. In such situations, it may be acceptable to wipe the outside of the article or other parts of the equipment where any leaked contamination is likely to accumulate.

A pass/fail criterion should be specified and the analysis technique should be sufficiently sensitive to record values below the pass/fail criterion. Records of leak tests should be maintained and included with the source records. Further advice on leak testing of sources can be found in ISO 9978: *Sealed Radioactive Sources – Leakage Test Methods*.

SUMMARY OF KEY POINTS

Identification of nuclides: α, β or γ spectrometry, β absorption or half-life measurements.

Determination of sample activity: counting efficiency, dependence on energy. Correction for counter background.

Efficiency of a counting system: the fraction of particles counted compared with the total number emitted.

Resolving time: detector is inoperative for short time after registering a pulse; this reduces effective counting time and is important at high count rates.

Counting statistics: standard deviation (σ); a measure of the accuracy of a count. $\sigma = \sqrt{N}$. To achieve good accuracy, counts must be large; 10 000 counts gives 1 per cent accuracy. Background count has an effect on accuracy.

Surface contamination level:

$$\text{Surface contamination level (Bq/cm}^2) = C_c \times \frac{100}{E_c} \times \frac{1}{A} \times \frac{100}{E_F}$$

Airborne contamination level:

$$\text{Contamination level (Bq/m}^3) = C_c \times \frac{100}{E_c} \times \frac{1}{V}$$

Leak test: a wipe test carried out at regular intervals on sealed sources to ensure that there is no significant leakage from the source.

REVISION QUESTIONS

1. Why is it usually necessary to determine the composition of the activity on an air sample filter paper? How can the nuclides be identified?
2. A sample gives a count of 16 347 in 1 min when counted in a Geiger–Müller counter of background 750 counts in 10 min, dead time 300 μs and efficiency 15 per cent. Calculate the sample activity.
3. What is meant by the resolving time of a detector and what is its importance in counting measurements?
4. The results of β absorption and decay measurements on a sample are shown below. Estimate the half-life and the maximum β energy. Refer to nuclide data and attempt to identify the nuclide present in the sample.

   Beta-absorption measurements:

Absorber (g/cm^2)	0	0.017	0.041	0.067	0.094	0.120	0.143	0.168
Corrected counts/min	3613	3324	2867	2310	1897	1563	1306	1097

   Decay measurements:

Time (days)	0	1	3	6	9	12	15	18
Corrected counts/min	3613	3376	3136	2637	2353	1980	1768	1510

5. What is the significance of the standard deviation of a measurement and how is its value calculated for a single counting measurement?
6. A sample is counted in a β castle and registers 22 501 counts in 50 min. The background count gives 2040 in 30 min. Calculate the corrected count rate and the standard deviation.
7. What is the most appropriate method to leak test an easily accessible 100-kBq ^{90}Sr source?

11 Radiation protection in the nuclear industry

11.1 INTRODUCTION

The discovery of fission in 1938 provided the basis of a new source of energy that was potentially greater than the entire world's reserves of fossil fuels. The first fission reactor was operated in a converted squash court in Chicago in 1942 by Enrico Fermi, and this was followed by the rapid development of nuclear power plants in the 1960s and 1970s. This slowed down in the 1980s and 1990s, and many of the early plants are now being decommissioned. However, nuclear energy still supplies a significant portion of the power requirements of most of the advanced countries of the world.

The advantages of the nuclear reactor as a source of power are offset to some extent by a number of special problems. These include:

1. the protection of the operator and maintainer;
2. the safe treatment and disposal or storage of the radioactivity produced; and
3. the need to achieve an acceptably low risk of injury to the public from the potentially large releases of radioactivity that could occur in the event of a reactor accident.

None of these problems is insurmountable and it is generally agreed that, with a small number of exceptions discussed in Chapter 16, the nuclear energy industry has a very good safety record.

To understand the hazards associated with reactors, a basic knowledge of nuclear fission and reactor technology is necessary. In this chapter, after a basic discussion of the process of nuclear fission and of reactor technology, the radiological hazards involved in the operation of nuclear reactors are outlined. This is followed by a description of the nuclear fuel cycle and the associated radiological issues.

11.2 FISSION

11.2.1 The fission process

Fission is the splitting of a nucleus into two approximately equal parts known as **fission fragments**. Certain types of heavy nuclei, notably uranium and thorium, are found to undergo spontaneous fission at a rather low rate. Others can be made to fission by the addition of energy, for example by bombardment with neutrons. Materials which can be made to fission in this way are said to be fissile. The process of fission results in the release of energy, mainly in the form of kinetic energy of the fission fragments. This is rapidly

converted into thermal energy and raises the temperature of the fuel material. Of naturally occurring materials, only the isotope uranium-235 (^{235}U) is fissile to a significant extent. This constitutes only 0.7 per cent by weight of natural uranium, the remaining 99.3 per cent being uranium-238.

A fundamentally important feature of fission is that the fission fragments are so unstable that they give off neutrons, usually between two and four per fission (Fig. 11.1). Most of the neutrons are emitted almost instantaneously and are called prompt neutrons, but some are released some seconds or even minutes after fission; these are called delayed neutrons and have an important role in controlling the nuclear chain reaction.

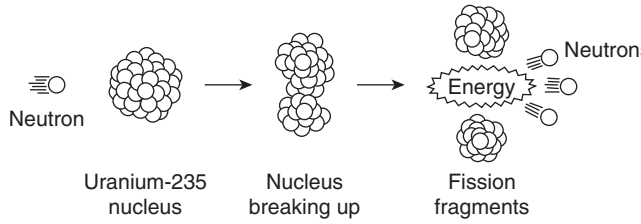


Figure 11.1 Fission.

The emission of neutrons by the fission process has a number of consequences. First, it makes possible a chain reaction; second, neutron capture reactions in the uranium fuel result in the production of transuranic elements, including plutonium (see section 11.2.4); and, third, neutron captures in the structural and other materials of nuclear reactor plants result in these materials becoming activated (see section 11.2.5).

11.2.2 Chain reactions and criticality

The neutrons emitted after fission may themselves cause further fission with the emission of more neutrons, thus making possible **a chain reaction**. In practice, some of the neutrons escape from the system and others are captured in non-fission reactions. In a reactor, wastage of neutrons by capture reactions is minimized by taking care to exclude materials of high-capture cross-section (i.e. materials that are strong absorbers of neutrons) from the core. Some loss is inevitable since even the fuel material can capture neutrons without fission. By increasing the size of the core, the fraction of neutrons escaping can be reduced to a sufficiently low level to permit a chain reaction. If a core is very small, neutrons can easily escape, but in larger cores they would be required to travel further and are, therefore, more likely to cause fission. Another method of reducing leakage is to put a reflector of some light material around the core to reflect escaping neutrons back into the core.

Consider an assembly of fissile material in which, on average, 2.3 neutrons are produced by each fission. If, on average, 1.3 neutrons are lost by leakage or capture, one neutron is available to cause a further fission. This results in a self-sustaining chain reaction in which the fission rate is constant. Such a system is said to **be critical**. An average loss of 1.31 neutrons for every fission means that only $2.3 - 1.31 = 0.99$ neutrons are available for fission. The system is then subcritical and the fission rate will decrease. Conversely, for a loss of 1.29 neutrons per fission, 1.01 neutrons remain. This results in a supercritical system in which the fission rate will increase.

Thus, for any given type of reactor there is a minimum size of core below which the system cannot go critical. Alternatively, a certain minimum mass of fuel is required for a chain reaction to be possible; this is the minimum **critical mass**. For example, a nuclear weapon can, in principle, be made from two pieces of fissile material, each of which is slightly more than one-half of the critical mass. The weapon is detonated by bringing the two pieces rapidly together to give a supercritical mass. This results in the fission rate, and hence the rate of energy release, rising very rapidly to enormous values. Fortunately, this is rather difficult to achieve in practice.

The possibility that fissile material could 'go critical' if assembled in a sufficient quantity has very important implications in the design and operation of plants in which it is processed, handled or stored. It is obviously very important to ensure that criticality does not occur, and this is usually achieved by careful design of the facilities supplemented, in some cases, by operational procedures. This aspect is discussed further in section 16.5.

11.2.3 Fission products

In fission, the splitting of the atoms of a fissile material can occur in many different ways. The most likely division of the heavy atom is into fragments of mass about 97 and 135. For example,

$$^{235}_{92}\mathrm{U} + {}^{1}_{0}\mathrm{n} \longrightarrow {}^{135}_{52}\mathrm{Te} + {}^{97}_{40}\mathrm{Zr} + 4\,{}^{1}_{0}\mathrm{n}$$

The distribution of mass number of fission products is of the form shown in Figure 11.2. The fission process produces about 300 different nuclides, most of which are rich in neutrons and decay by a series of β emissions through a decay chain of radioactive nuclides. In the example of fission shown above, tellurium-135 and zirconium-97 both undergo a series of β decays until a stable nuclide is produced:

$$^{135}_{52}\mathrm{Te} \longrightarrow {}^{135}_{53}\mathrm{I} \longrightarrow {}^{135}_{54}\mathrm{Xe} \longrightarrow {}^{135}_{55}\mathrm{Cs} \longrightarrow {}^{135}_{56}\mathrm{Ba}$$

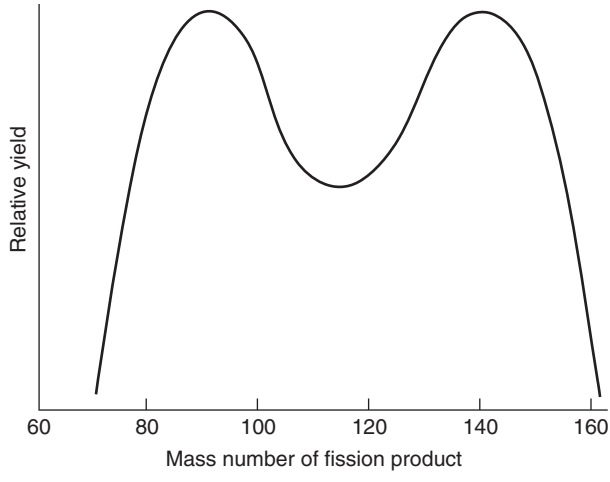


Figure 11.2 Relative fission product yield.

and:

$$^{97}_{40}\text{Zr} \longrightarrow {}^{97}_{41}\text{Nb} \longrightarrow {}^{97}_{42}\text{Mo}$$

It will be seen that the utilization of nuclear fission to produce energy results in the formation, within the fuel, of hundreds of different types of radioactive fission products, with half-lives varying from a fraction of a second to very many years. The inventory of fission products in the fuel builds up over the period of irradiation.

Figure 11.3 shows the fission product inventory in becquerels, and its decay from 1 to 1000 days in fuel, which has produced a uniform 1 MW of heat over a 2000-day irradiation.

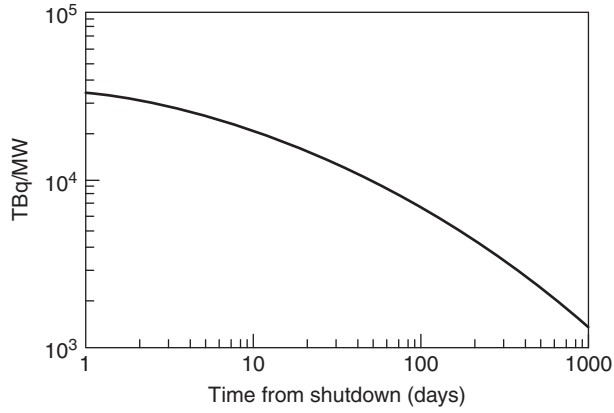


Figure 11.3 Approximate fission-product β inventory in fuel, irradiated for 2000 days at 1 MW.

The many different nuclides present as fission products have a very wide range of physical and chemical properties, and so the radiotoxicity of the different nuclides varies enormously. To prevent the fission products escaping, the fuel is contained within another material of suitable properties, known as the **fuel cladding**.

11.2.4 Transuranic elements

As previously noted, neutrons can be captured by fuel atoms in non-fission reactions. Successive capture reactions and β decays result in the generation in the fuel material of **transuranic elements**, that is, elements higher in the periodic table than uranium. Of particular importance is neutron capture in ^{238}U, yielding ^{239}U, which quickly undergoes β decay to neptunium-239 (^{239}Np) and then to plutonium-239 (^{239}Pu).

$$^{238}_{92}\text{U}(\text{n}, \gamma)^{239}_{92}\text{U} \xrightarrow{\beta} {}^{239}_{93}\text{Np} \xrightarrow{\beta} {}^{239}_{94}\text{Pu}$$

This is a process of great importance since it provides a route whereby a large proportion of ^{238}U, which is not fissile to a significant extent, can be converted into ^{239}Pu, which is fissile. However, ^{239}Pu has the disadvantage of being a long-lived ($T_{½}$ = 24 300 years) α-emitter and, unlike uranium, it is of very high radiotoxicity.

Further capture and β-decay processes lead to the production of americium (Am) and curium (Cm) nuclides.

11.2.5 Activation products

Neutron capture in structural materials and in the coolant of a reactor results in the generation of many species of radioactivation products. In steel components, the stable iron isotope ^{58}Fe undergoes neutron capture to become ^{59}Fe, which decays by β emission to cobalt-59 (^{59}Co) with a half-life of 45 days.

In many cases, because of a particular combination of nuclear properties, activation of trace elements in a material is important. For example, all types of steel contain cobalt, normally at a concentration of only a few hundred parts per million. The stable isotope ^{59}Co is activated by the reaction:

$$^{59}Co\ (n, \gamma)^{60}Co$$

Cobalt-60 has a relatively long half-life of 5.22 years and, in addition to a low-energy β particle, emits two energetic γ-rays per disintegration. Another important activation product, particularly during decommissioning, is nickel-63 (^{63}Ni). Nickel-63 is formed by the reaction:

$$^{62}Ni\ (n, \gamma)\ ^{63}Ni$$

Nickel-63 has a half-life of 96 years and decays by β emission (E_{max} 0.067 MeV).

In both water-cooled and gas-cooled reactors, an important reaction on the oxygen component of the coolant is:

$$^{16}O(n, p)^{16}N$$

Although nitrogen-16 has a very short half-life of 7.2 s, it has a great influence on the shielding requirements for all reactors.

11.3 REACTOR SYSTEMS

11.3.1 General features

The great majority of commercial nuclear power reactors currently operating or under construction worldwide are light water reactors (LWRs), which use water as both a coolant and a neutron moderator (see section 11.3.2). These are of two types, **pressurized water reactors (PWRs)** and **boiling water reactors (BWRs)**. In a PWR, a closed water-coolant system transfers heat from the core to heat exchangers, which raise steam in a secondary circuit to drive a turbogenerator (see Fig. 11.4). Bulk boiling of the water in a PWR is prevented by maintaining the system at a very high pressure (about 2200 psi or 150 bar). In a BWR, however, the pressure of the system is such that the water boils and the resulting steam, after passing through a steam separator, passes directly to the turbines. Typical PWR and BWR plants are illustrated in Figures 11.5 and 11.6. Another type of water-cooled reactor is the Canadian CANDU system, which uses heavy water as the coolant and neutron moderator.

In the UK, a different line of development was pursued which resulted in two generations of **gas-cooled reactors**, the Magnox and the advanced gas-cooled reactors (AGRs). The Magnox reactor design is now obsolete and most of the stations have been

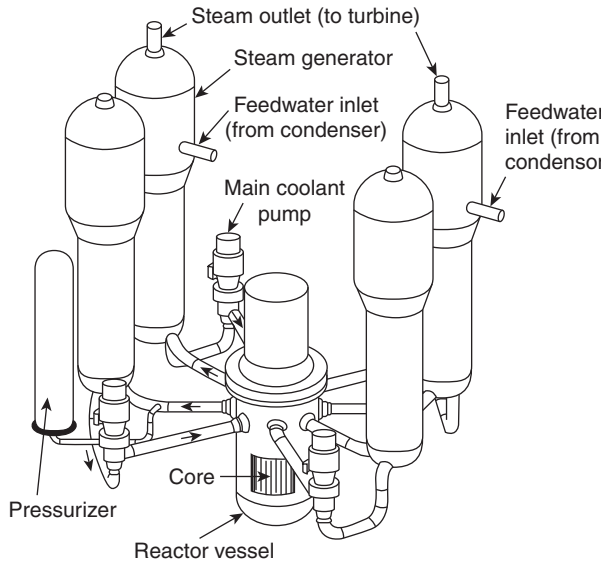


Figure 11.4 Primary system of a pressurized water reactor.

shut down. Most of the second-generation AGR plants, which use stainless steel fuel cladding and enriched uranium, are still operating, but they will all be shut down over the next decade or so. Another reactor system is the **sodium-cooled fast-breeder reactor**, sometimes known as the **liquid-metal fast-breeder reactor** (**LMFBR**). The original concept was that fast reactors would use plutonium fuel extracted from spent thermal reactor fuel but, later, would breed enough plutonium by neutron capture in ^{238}U in and around the core to meet their own fuel requirements. The development of fast reactors has slowed down considerably because of difficult engineering problems leading to significantly increased cost, and also because the expected scarcity of uranium has not materialized.

There has been some interest over the years in the **high-temperature gas-cooled reactor** (**HTGR**), which uses helium gas to cool ceramic uranium fuel. In one design, known as the **pebble bed reactor**, the fuel consists of many thousands of ceramic spheres, through which the helium passes to remove the heat. The temperature of the helium gas as it exits the core is much higher than in existing gas-cooled reactors and so a direct turbine cycle should theoretically be possible. However, the HTGR presents a number of novel engineering and radiological challenges which have so far prevented its commercial exploitation.

The central features of any nuclear reactor are the core, which contains the fissile material, a means of controlling the fission rate and a moderator in the case of thermal reactors but not in the case of fast reactors. The other main features are a cooling system to remove the heat generated by fission and a radiation shield, often called the biological shield. These various features are described below.

11.3.2 The core and control system

The **reactor core** contains fuel assemblies or elements that consist of fissile material in a fuel can or covered by a cladding material, which prevents the escape of the fission

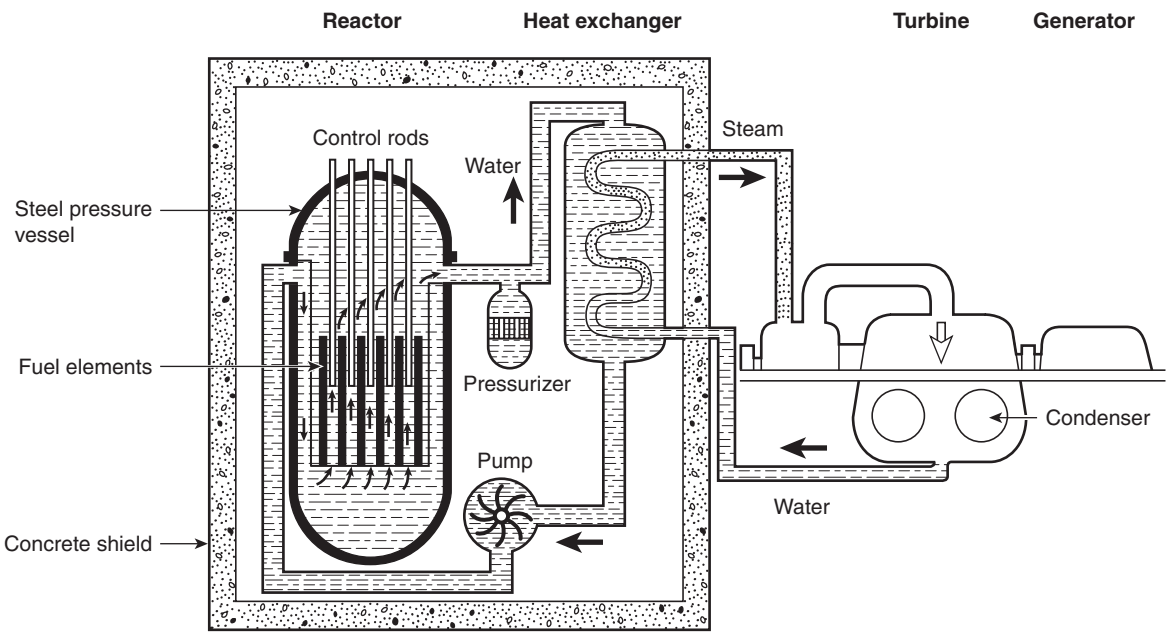


Figure 11.5 Schematic illustration of a pressurized water reactor.

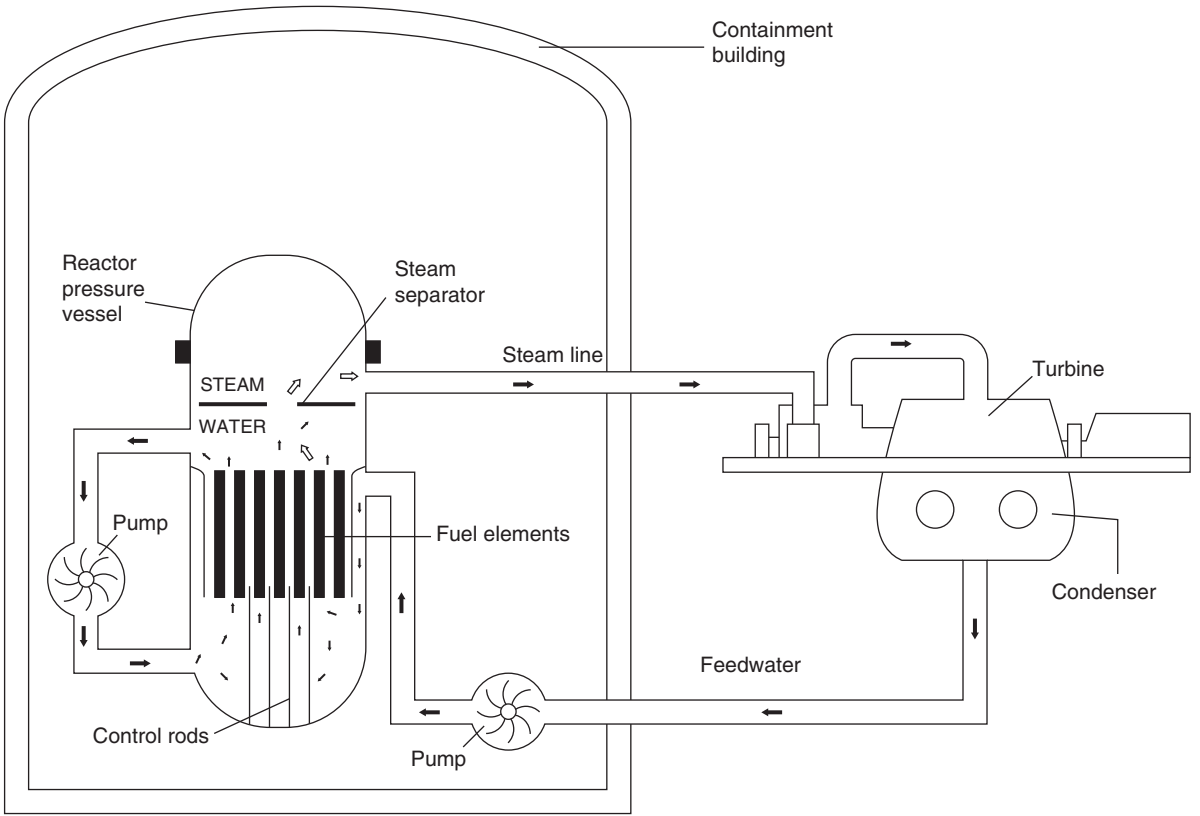


Figure 11.6 Schematic illustration of a boiling water reactor.

products. The fuel is normally uranium, although plutonium and thorium are sometimes used. Naturally occurring uranium contains 0.7 per cent of the isotope ^{235}U and 99.3 per cent of ^{238}U. Magnox and CANDU systems are able to operate using natural uranium, but in other systems it is necessary to use uranium containing a higher percentage of ^{235}U, and this is achieved by means of an enrichment process.

The **moderator** serves to slow down the fast neutrons produced by fission to the thermal energy range in which they are most likely to cause further fission. In water-cooled reactor systems, the water serves as both the moderator and the coolant, while in the gas-cooled systems the moderator is graphite.

The **control rods** are made from materials of high neutron-absorption cross-section such as boron or cadmium. Withdrawal of the control rods to a certain minimum position takes the reactor critical. Further withdrawal makes the reactor supercritical and the fission rate increases. When the required power is reached, insertion of the rods to the critical position causes the power to become constant. The reactor power can be reduced by inserting the rods to a subcritical position. The control system is based on a number of neutron detectors around the core. If the neutron flux exceeds a preset value or if it is increasing too rapidly, the detectors sense it and the reactor is automatically shut down by the rapid insertion of the control rods (or sometimes special shut-down rods). Various other monitoring systems provide data which can shut down the reactor if something goes wrong, for example by detecting high γ-radiation levels outside the biological shield, excessive core or coolant temperature, or loss of coolant flow.

11.3.3 The cooling system

Fission causes energy release within the fuel and consequent temperature rise of the fuel and cladding material. The heat is removed by the **coolant** and steam is raised, directly or indirectly, to drive turbines. Various subsidiary systems are provided for coolant purification, sampling and safety purposes.

The integrity of the cooling system must be very high. Most coolants become radioactive to some extent and even small leaks may pose radioactive contamination problems. Large leaks could limit the heat removal from the core and affect the safety of the reactor. It is important to realize that in a power reactor, the decay of the fission products generates a substantial amount of heat, and so cooling must be guaranteed even when the reactor is shut down.

11.3.4 The biological shield

The purpose of the **biological shield** is to attenuate the neutron and γ radiation from the reactor core and cooling system so that the operators and maintenance staff do not receive excessive doses. The most commonly used materials are lead, concrete, iron, water and polythene. In the case of power reactors, the factor of reduction required may be of the order of 10^8–10^9. Weaknesses can arise in the shield where coolant ducts and other penetrations offer streaming paths for radiation. The shield is often constructed in two parts: the primary shield around the reactor core and the secondary shield around the cooling system. This arrangement permits access to the coolant system (circulators, heat exchangers, etc.) when the reactor is shut down since the γ radiation from the decay of fission products is attenuated by the primary shield.

11.4 REFUELLING REACTORS

In a nuclear reactor, 1 kg of natural uranium (occupying a volume of about 50 cm^3) can produce as much energy as 10 tonnes of coal burned in a conventional power station. This is despite the fact that only 0.3 per cent of the uranium atoms are burned up, which represents about 40 per cent of the fissile ^{235}U atoms. The percentage of uranium that can be burned up is limited, since a stage is eventually reached when insufficient fissile material remains to sustain the chain reaction. This is exacerbated by the build-up of fission product poisons, that is, long-lived neutron-absorbing fission products, in the fuel. In addition, pressure generated by the fission products in the fuel causes swelling and distortion of the fuel elements. When a fuel element reaches the required burn-up, it must be removed from the reactor (remember that it is intensely radioactive) and replaced by a fresh element.

In both PWRs and BWRs, **refuelling** is performed during shut-down at intervals of 1–2 years, when about one-quarter to one-third of the core is replaced. This involves flooding a canal above the reactor pressure vessel, removing the pressure vessel lid and lifting the fuel assemblies out one at a time into the water-filled cavity. They are then transferred down the canal into a storage pond.

In other reactor designs, such as CANDU, RBMK and AGRs, refuelling can be performed with the reactor operating. This is termed **on-load refuelling** and is intended to improve the load factor and economics of the nuclear power plant. For example, AGR refuelling is performed by a remotely controlled charge/discharge machine on top of the reactor, which removes the shielding plug, seals itself onto the pressure vessel and removes the fuel channel cap. The spent fuel elements are raised from the channel into the machine, which is heavily shielded, and replaced by fresh elements. When the machine contains its full quota of elements, it is moved to a position where the elements can be released through a shielded chute into a large tank of water, known as a cooling pond, which provides shielding and cooling for the spent elements. The machine is designed so that these operations can be carried out safely. It must, therefore, provide shielding from the reactor core above the unplugged hole, provide shielding from the spent fuel elements inside the machine, prevent leakage of gas or contamination and provide cooling for the fuel elements (the heat generated by decay of the fission products in an element when it is removed from the reactor is sufficient to cause it to overheat unless external cooling is provided). For various technical reasons, AGR on-load refuelling is now carried out only at reduced power.

11.5 RADIATION HAZARDS FROM REACTORS

11.5.1 General

In general, reactors are a lesser radiation hazard when they are operating than when they are shut down. The shielding is designed to give acceptable radiation levels at working positions and this is confirmed by thorough surveys during the commissioning of the plant and subsequently at regular intervals. Systems are provided which permit safe means of sampling coolants and other radioactive effluents. During shut-down periods, a great variety of non-routine jobs may be undertaken, some of them on highly radioactive systems. It is during such periods that exposure of personnel to radiation and radioactive contamination must be carefully controlled.

A serious fault or maloperation could cause considerable damage to the plant and give rise to dangerously high levels of radiation or radioactive contamination. If the hazard is confined to the reactor site, it is often called a site emergency, but if it extends off site then it may become a public emergency. Accidents of this type are discussed in Chapter 16.

11.5.2 Sources of radiation

The main sources of radiation from a reactor at power are the core and the coolant. The radiation from the core includes fission neutrons, fission γ-rays, fission product decay γ-rays, neutron capture γ-rays and activation product decay γ-rays. The last two arise predominantly in the core structure and shield. The radiation from the coolant is mainly γ-rays arising from activation of the coolant, activation of impurities and fission-product contamination of the coolant. The sources are illustrated in Figure 11.7.

Radiation from the core

The neutrons produced in a reactor as a result of the fission process are fast neutrons in the range of 0.1–15 MeV, with an average energy of about 2 MeV. Those emerging from the surface of the biological shield have undergone varying degrees of moderation and so neutrons of all energies from thermal to fast may be present.

Fission γ-rays are those emitted immediately after the fission fragments and vary in energy from 0.25 to about 7 MeV. The γ radiation resulting from the decay of fission products in the fuel elements is small compared with the fission γ radiation but, whereas the latter ceases on shut-down of the reactor, the fission products continue to emit radiation for many years after the fuel has been withdrawn from the reactor.

Neutron capture in the structural materials of the reactor and in the shield results in the emission of capture γ-rays and makes these materials radioactive. The radiation from the decay of the radioactivity, as in the case of fission products, continues to be emitted

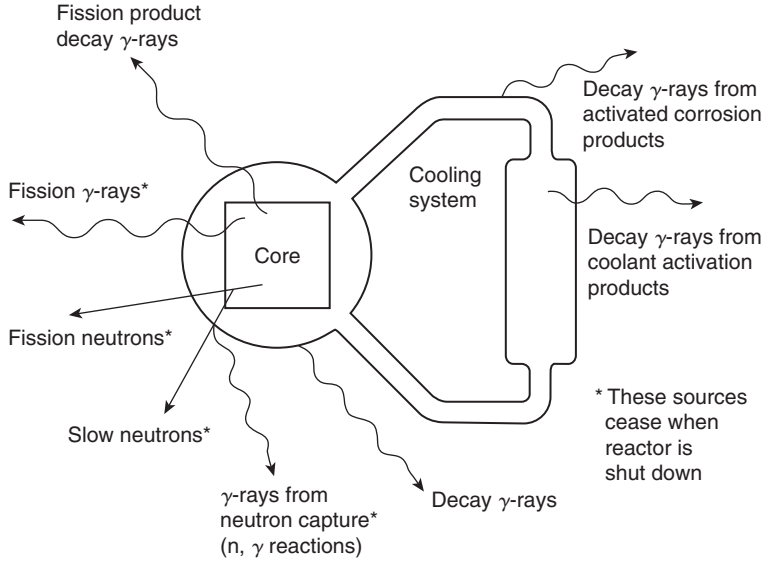


Figure 11.7 Sources of radiation in a nuclear reactor system.

when the reactor is shut down. Neutron scattering also leads to γ-ray emission but does not, in general, induce radioactivity.

Radiation from the coolant

As noted previously, in both water-cooled and carbon dioxide-cooled reactors, the reaction

$$^{16}O(n, p)^{16}N$$

is important because nitrogen-16 decays with a half-life of 7.2 s and emits very penetrating (6.1-MeV) γ-rays. This means that the coolant circuit of the reactor is a significant source of γ radiation and, in most cases, must be shielded.

Air is present as an impurity in the coolants of both gas- and water-cooled reactors. Argon, which is present to the extent of 1.3 per cent in the air, is activated by neutrons via the reaction:

$$^{40}Ar(n, \gamma)^{41}Ar$$

Argon-41 decays with a half-life of 1.8 h and emits γ radiation with an energy of 1.29 MeV. It can contribute to operator dose during online sampling and immediately after reactor shutdown.

In LMFBRs, which use sodium as a coolant, the important activation product is sodium-24 (^{24}Na), which is produced by an (n, γ) reaction on sodium-23.

Although reactor coolants are very pure by normal standards, they always contain some **impurities**. When subject to the very high neutron flux in the reactor core, these impurities become radioactive to an appreciable extent. Impurities also arise as a result of corrosion or erosion from the core and the walls of the coolant system. In water-cooled reactors, iron, nickel, cobalt and manganese are common impurities because of corrosion of the coolant system. Deposition of this corrosion material in the core causes the build-up of a film of corrosion products on fuel surfaces. As a result of irradiation by neutrons, these films become highly radioactive. The continuous release of material from the core and subsequent deposition in the out-of-core regions causes a build-up of radioactivity in the coolant system which is known as **crud** in PWR systems. In Magnox and AGRs, graphite dust (which itself contains impurities) collects around the cooling system.

Reactor coolants usually contain readily detectable levels of fission-product contamination arising from:

1. uranium contamination on the fuel element surface;
2. uranium impurity in the fuel cladding material; and
3. release from any damaged fuel elements.

The coolant is being continually cleaned up by the coolant treatment system and so the long-lived fission products do not build up appreciably. The predominant fission product activities are usually krypton-88 (^{88}Kr) and xenon-138 (^{138}Xe), which are inert gases, their particulate daughter products rubidium-88 (^{88}Rb) and caesium-138 (^{138}Cs), and the three isotopes of iodine, ^{131}I, ^{133}I and ^{135}I. A seriously damaged fuel element could lead to considerable fission product activity being spread around the cooling system. In water-cooled reactors, the presence of fission products in the coolant is detected by sampling and radiochemical analysis. This is carried out routinely while the reactor is at power using

special sampling facilities. The samples are tested for the presence of fission products, including nuclides of iodine, caesium and strontium. Most gas-cooled reactors are fitted with a system known as '**burst can detection**', which continuously and automatically 'sniffs' each channel in turn and gives warning to the operators if an increase in the fission product activity should occur. The channel containing the damaged fuel is then unloaded and the faulty element is replaced.

11.5.3 Sources of radioactive contamination

Beta emitters

Almost all of the radioactive nuclides mentioned in the preceding paragraphs decay by β emission. **Beta radiation** is so easily absorbed that the shielding designer does not even need to consider it, concentrating instead on the associated γ emission. On the other hand, if radioactive contamination occurs because of a leak of radioactivity from the reactor system, the β radiation is often of prime importance.

A radionuclide of considerable importance that is produced in all reactor systems is tritium (^{3}H). This nuclide has a half-life of 12.3 years and decays by low-energy β emission only. It is produced by fission, by an (n, γ) reaction on deuterium (hydrogen-2) and by various reactions on lithium and boron. In reactors cooled or moderated by heavy water, large amounts of tritium build up. In light water systems, because of the much lower concentration of deuterium and the frequency of water change (heavy water is much too expensive to change), build-up of tritium is usually less significant. Lithium and boron are present in most systems, either as additives, neutron absorbers or impurities, and can contribute to tritium production by a number of reactions, including:

$$^{6}Li(n, \alpha)^{3}H \text{ and } ^{10}B(n, 2\alpha)^{3}H$$

Coolant leaks

Contamination can, of course, occur because of a **coolant leak**. In pressurized water systems, the leak may be direct to the atmosphere or via a heat exchanger into the secondary system. In the latter case, radioactivity, mainly the gaseous activities ^{88}Kr, ^{138}Xe and ^{41}Ar, would be carried over with the steam into the turbines and then to the atmosphere via the condenser air ejector. As noted above, the fission product gases ^{88}Kr and ^{138}Xe decay to their particulate daughters ^{88}Rb and ^{138}Cs. Note that a leak in the heat exchanger of a gas-cooled reactor would normally cause steam to leak into the primary system because of the higher secondary pressure. It should also be noted that, in a BWR, the steam from the reactor carries with it ^{16}N from the reactor water into the turbines and associated plant. Even though its half-life is only 7.2 s, the high-energy γ radiation results in significant radiation levels in the vicinity of the steam systems which require the provision of shielding around the steam pipes, turbine and other major items.

Containment

The core of a reactor at power contains about 0.2 TBq of fission products per watt of thermal power. Thus a reactor operating at, say, 1000 MW contains about 2×10^{8} TBq of fission products. This vast inventory of radioactivity is contained within the fuel can or cladding, which provides the first level of **containment**. The second level of containment is the boundary of the primary systems, that is, the pressure vessel and the coolant system.

This boundary also contains the radioactivity of the coolant which, in a 1000-MW reactor, may amount to some hundreds of terabecquerels. Provided that the primary and secondary containments remain intact, there is little risk of serious contamination. In practice, some contamination does occur during the operation of most reactors. For example, it is usually necessary to sample the coolant periodically and there is often some radioactive effluent. A contamination hazard could arise in both cases but the risk is minimized by good design of facilities.

In its most common usage, the term 'reactor containment' refers to the structure within which the whole system is housed. The main function of this containment is to protect the general public by limiting the release of fission products in the event of a serious reactor accident, although its secondary function, to protect the reactor systems from external hazards and terrorist threats, has become increasingly important over the past decade or so (see section 16.4).

11.5.4 The shut-down reactor

Maintenance

A reactor represents a large capital investment, and **shut-down periods**, whether scheduled or not, are costly. Reactor systems are comparatively simple, well engineered and normally very reliable. The majority of maintenance, either corrective or preventative, is on ancillary or secondary equipment. A major overhaul may include decontamination and refitting of coolant circulators, control rod mechanisms, inspection of heat exchangers and various other jobs on radioactive systems. At such times, the need to keep to a tight schedule can lead to a general reduction in standards of safety, both radiation and conventional, because it may slow down the work. To prevent this state of affairs, all major work must be planned in consultation with interested parties and sometimes it is desirable to make a mock-up to allow particularly difficult jobs to be rehearsed under non-active conditions. Personnel should receive instruction in general safety matters and be familiar with the particular hazards associated with their own work.

External radiation

When the reactor is shut down, the primary shield gives adequate protection against the fission products in the core. The radiation hazard to personnel working on the primary system is caused by radioactivity within the system. The **dose rate** in the vicinity of the primary system tends to decay rapidly in the first 24 h after shut-down, mainly because of the decay of coolant activities or their clean-up by the treatment system. Thereafter, the levels do not change significantly from day to day. The half-lives of most of the radioactive corrosion products are in the range of 1 month to about 5 years. The dose rates vary considerably from reactor to reactor but, in systems with corrosion problems, levels of 10–100 mSv/h can be encountered on certain components. If the dose rate is excessive, it is sometimes possible to provide additional shielding on 'hotspots'. An alternative approach is to decontaminate the component, but this would be done only during major shut-down periods.

Careful control is required of personnel working in areas of high dose rate. This often takes the form of a manned control point at the entrance to the area. Personnel entering the area are given a 'working time' based on a radiation survey of the area. In addition to their normal personal dosimeter they are required to wear some form of direct reading device

such as an electronic dosimeter. The times of entry and exit and the dosimeter reading are logged. Some electronic dosimeter systems can be read and logged automatically and remotely.

Contamination

As noted earlier, reactor coolants normally contain measurable amounts of radioactive fission products and activated corrosion products. **Contamination** is likely to occur during maintenance operations that involve breaching the primary coolant system. It is obviously essential to depressurize the primary circuit before attempting to breach it, and it is general good practice for the personnel involved to wear full face masks when first breaking into any part of the system.

The standards of protective clothing required on a particular plant are evaluated from experience. As well as the presence of contamination, there are often other factors such as temperature, humidity and the possible presence of toxic gases which affect the choice of protective clothing and equipment. Personnel cannot be expected to wear impervious clothing, such as PVC suits, in temperatures of 40–50°C unless the suits are fully ventilated. This in turn causes difficulties in confined spaces because of the required air lines. These and other considerations mean that all maintenance operations on the primary circuit must be planned and executed carefully. It is essential that the necessary changing, monitoring and data-logging facilities are established before the maintenance activity begins. If significant levels of contamination are present, there should be an attendant to assist in the removal of contaminated clothing. Frequent monitoring of levels of contamination both inside and outside the area should be undertaken to ensure proper control.

11.6 RESEARCH REACTORS

Research reactors, of which there are many different types, present special health physics problems. The reactors have a wide range of applications, including fundamental research, materials testing and the commercial production of radioactive sources. The main radiological problems usually arise not from the reactor but from the experimental equipment. For example, there are often holes through the shielding to obtain high-intensity beams of neutrons outside the reactor for various purposes. Careful consideration needs to be given to the monitoring and shielding requirements for such high-intensity beams, recognizing the special properties of neutrons (as discussed in section 8.5). Rigs containing experimental equipment or materials become highly radioactive. In these and other situations, special precautions are necessary to protect both operators and users of the reactor.

11.7 FUEL STORAGE PONDS

11.7.1 Introduction

About one-quarter to one-third of the total fuel load of a reactor is removed from the core every 12–18 months and is replaced with fresh fuel. Spent fuel rods generate considerable heat and intense radiation. After removal from the reactor, it is usual to allow the fuel to decay for some months or years in a fuel cooling (or spent fuel) pond situated close to the reactor. In this **pond**, the water cools the fuel and provides shielding against the radiation emitted. Such storage eases some of the problems involved in moving the highly

radioactive fuel from the reactor site to the fuel reprocessing plant or storage facility. In addition to the routine operational problems, ponds pose two special hazards, criticality and loss of shielding accidents, which are discussed below.

11.7.2 Criticality

Ponds often contain enough fissile material to 'go critical', that is, to initiate a fission chain reaction. In general, if only natural uranium fuel is in the pond, **criticality** is not possible even under the worst possible conditions. If enriched uranium or another fissile material such as plutonium is present, criticality could conceivably occur. The hazard is controlled by storage in specially designed metal racks which ensure safe configurations, that is, with adequate spacing between fuel elements. Usually there is also a limit on the number of elements that may be out of their storage position at any one time.

11.7.3 Loss of shielding

A typical storage pond may often contain thousands of terabecquerels of fission product activity and a single element may contain a few hundred terabecquerels. Unshielded, such an element would give a γ dose rate in excess of 1 Sv/h at a distance of a few metres. Obviously, this situation must not be allowed to occur. **Loss of shielding** can occur through loss of pond water or in a fuel-handling accident, in which a fuel element is raised too close to the surface of the water. Loss of pond water can potentially result either from accidental pumping out or from a leak caused by serious structural damage. The possibility of pumping out can be minimized by good design of the water system and by administrative control of the system (e.g. locks on vital valves). Serious structural damage is less likely, although the possibility of fuel storage ponds being the target of terrorist activity has led to a much closer scrutiny, over the past decade or so, of their ability to withstand significant external challenges.

Fuel-handling accidents are difficult to safeguard against completely without defeating the advantages of water shielding, such as visibility and flexibility. A good design of lifting tackle can make it impossible to raise fuel too high, but only if the correct tackle is used. A little ingenuity can defeat the best systems. Considerable reliance is placed on installed γ monitors to give warning of potentially dangerous situations.

11.7.4 Operational aspects

The fission product activity of the fuel is contained within the fuel cladding. A breach of the cladding will lead to contamination of the water by fission products, the extent of which will depend on the size of the leak and the form of the fuel. In order to detect the release of gaseous fission products, which would be rapidly released from the water surface, many ponds have an **air extraction system** 'sweeping' the water surface. Some degree of water contamination also occurs as a result of the release of radioactive corrosion products from the surface of the fuel cladding. To prevent a build-up of activity in the pond, a water treatment system is provided. A water cooler is usually necessary depending on the amount of fuel stored.

11.7.5 Pond instrumentation

There are three main types of instrument used to give warning of hazardous conditions:

1. Installed **γ monitors** are an essential feature of any fuel storage pond. At least three instruments are necessary and they should be situated so that they cannot

be shielded from the pond surface. A local alarm such as a bell or buzzer should sound if any of the instruments reaches some quite low level, say 0.1 mSv/h. This would warn the operators that a potentially dangerous situation is developing. At some higher radiation level, say 10 mSv/h at the **operator's position**, an evacuation alarm should sound. The pond alarm system should be designed so as to avoid spurious evacuations. The logic should take account of the positions of the alarms as well as their number and magnitude. In a properly designed alarm system, an evacuation may be triggered either by one high-level alarm or by several adjacent lower-level alarms.

2. A **pond water counter** is a simple device which continuously monitors the water for β activity and gives early warning of damaged fuel. A shielded liquid-flow Geiger–Müller tube connected to a ratemeter is often used.
3. An **air monitor** is a continuously operating airborne particle monitor in the general area or in the air extract that gives warning of high airborne contamination arising from pond operations or maintenance of pond equipment.

11.8 THE NUCLEAR FUEL CYCLE

The complete nuclear fuel cycle involves a number of 'front-end' operations, including mining, extraction of uranium, uranium enrichment and fuel fabrication to produce the fuel for nuclear reactors. After removal of the irradiated fuel from the reactor, various 'back-end' activities are required to ensure the safe management of the fuel and associated wastes.

11.8.1 Front-end operations

The front-end operations comprise the mining, milling, extraction and enrichment of uranium and its fabrication into fuel elements or assemblies for loading into nuclear reactors.

Uranium mining

The mining of uranium ore is undertaken using both underground mining and open-pit working. In most cases, the level of radioactivity of the ore is quite low and the main radiological issue is the airborne concentration of radon and its decay products in the mine atmosphere. This is controlled to a large extent by achieving a high ventilation flow in the mine workings. After mining, the uranium ore is crushed into a fine powder (usually referred to as milling) and is followed by a chemical leaching process which produces a precipitate, known as **yellowcake**, which contains various uranium compounds, particularly U_3O_8. The radiation risk to workers is generally quite low and the main issue is that the waste products, known as **tailings**, contain most of the radioactive decay products of uranium, including radium and its decay product, radon. These require careful management. One approach is to cover the waste material with several metres of top soil to limit both direct radiation levels and to reduce radon emanation. Another approach is to return the material underground. The yellowcake uranium oxide product is a material of low radiological impact and it can be safely transported in standard 200-L steel drums.

Radiation doses to underground mine workers are typically of the order of 10 mSv per year, of which about 30 per cent is from external radiation and the balance from inhalation of radon and its decay products. For open-pit operations, the annual dose to workers is lower, typically 2–3 mSv, and is mostly attributable to external exposure.

Uranium enrichment

Most nuclear reactors require fuel in which the concentration of the ^{235}U isotope is enriched from its natural level of 0.7 per cent to about 3.5 per cent. The enrichment process requires the uranium to be in a gaseous form, and this is achieved by converting the uranium oxide into uranium hexafluoride, UF_6, which is solid at room temperature but becomes gaseous at a temperature of 57°C. In the early days of nuclear power, enrichment was achieved by a process of **gaseous diffusion** in which the molecules containing the lighter isotope, ^{235}U, diffused through a porous membrane slightly more easily than those containing ^{238}U. This process was repeated through a cascade of more than a thousand diffusion stages in order to achieve the required enrichment. For every tonne of natural material processed, about 130 kg of 3.5 per cent-enriched uranium was produced. The balance of 870 kg in the rejected stream contains uranium with a lower concentration of the ^{235}U isotope (around 0.3 per cent), and this is known as depleted uranium (DU).

Gaseous diffusion requires a high-energy input and, for this and other reasons, has been largely replaced by the more energy-efficient **gas centrifuge process**. In this process, the UF_6 gas is fed into a very high-speed centrifuge where, because of the mass difference between molecules, there is a very small increase in the concentration of the heavier ^{238}U at the outer part of the centrifuge tube and an equivalent increase in the ^{235}U towards the centre. As with diffusion, the process requires thousands of stages in order to achieve the required degree of enrichment.

The bulk of the byproduct from enrichment is depleted uranium, which has some uses in such things as tank armour, radiation shielding and ballast. However, the great majority of DU is kept in storage at the enrichment sites in the form UF_6. The main safety issue associated with enrichment plants and the handling of DU is the high chemical toxicity of the UF_6. The radiation doses associated with uranium enrichment are generally low, usually less than 1 mSv per year.

Fuel fabrication

At the **fuel fabrication** plant, UF_6 is converted into uranium dioxide (UO_2) powder. The powder is then fabricated into pellets, which are fired in a high-temperature sintering furnace to create ceramic pellets of enriched uranium. These pellets are ground to achieve a uniform pellet size and then stacked within a corrosion-resistant metal alloy can (either stainless steel or zirconium alloy in most modern reactor fuels) which is filled with helium and sealed to form a fuel pin. All of these operations are undertaken within a strict quality-assurance regime to ensure that the fuel is of a high and consistent quality. It should be noted that a typical PWR core consists of 193 fuel assemblies, each made up of 204 fuel pins, each of which in turn contains a stack of about 250 pellets.

The low specific activity of uranium means that it is not necessary to provide a high degree of containment for fuel fabrication, although it is usual to operate process areas at a negative air pressure in order to reduce the spread of contamination. Similarly, because of the low γ emission, radiation shielding of the process lines is not usually required. Doses to workers in fuel fabrication plants are generally quite low, typically around 1 mSv per year.

11.8.2 Back-end operations

The back-end operations are those associated with the management of the spent nuclear fuel and the associated wastes. In some countries, notably France, Japan, Russia and the UK, the policy is to reprocess spent nuclear fuel in order to recover plutonium and the unused portion of ^{235}U for further use and to separate the other actinides and fission products for disposal. Other countries have adopted a policy of long-term storage, keeping open the option either to reprocess the spent fuel eventually or dispose of it directly when appropriate facilities become available.

Fuel reprocessing

Spent fuel is transported to the reprocessing plant in large high-integrity steel containers that provide containment, shielding and heat dissipation. The fuel is then usually stored in a cooling pond for a further period, typically 5 years, to allow cooling and radioactive decay of fission products.

In a reprocessing plant, fuel is first mechanically dismantled and the fuel pins are sheared into lengths of a few centimetres, which then pass into a dissolver vessel. Here the uranium oxide fuel is dissolved out of the fuel pin sections by nitric acid. The resulting raffinate is directed into the process systems, where it is chemically separated into three main streams: unused uranium, plutonium and a highly active waste stream which contains almost all of the fission products and higher actinides such as americium and curium.

A reprocessing plant poses much greater radiological problems than a reactor because of the nature of the processes involved. Essentially, a **reprocessing plant** is a complex chemical facility, processing intensely radioactive solutions that require multi-level containment and that must be shielded from operating areas by a metre or more of concrete. Routine and special maintenance has to be carried out and, although facilities are provided to enable much of it to be done remotely, some contact maintenance is inevitable. Although the part of the plant requiring maintenance will be decontaminated, this is never 100 per cent effective and so maintenance work invariably has to be carried out in contaminated areas. Another issue is that relatively small leaks of the radioactive solutions into the shielded 'cells' of the plant can cause severe radiation and contamination problems. Decontamination of cell and equipment surfaces is very challenging and time consuming.

Of the product streams, the uranium stream does not present any significant radiological problem, but the plutonium stream demands extremely high standards of containment to prevent any leakage that could lead to a potentially severe contamination hazard. The highly active waste stream is routed initially into special storage tanks, where it remains for a few years until it is ready for vitrification. This is a process in which the highly active waste is mixed with glass, turning it into a solid form and reducing its volume to about one-third of its original amount. Vitrification allows the material to be stored safely in a form that should be suitable for long-term storage or eventual permanent disposal. In addition to the highly active waste streams, there are a number of subsidiary streams, the treatments of which depend on local conditions. The management of radioactive waste is covered in Chapter 12.

In the first-generation reprocessing plants, levels of operator exposure were often very high. More modern plants are designed to minimize operator doses, particularly through the provision of automation and remote-handling technologies. Average annual doses to workers are now only a few millisieverts.

Another important factor in the design and operation of fuel cycle facilities is the possibility of a criticality accident. Such an accident could result not only in high neutron and γ radiation from the event, but also, because of the energy generated, in dispersal of the process material.

Long-term storage

Various methods are in use for the long-term storage of spent fuel pending a decision to reprocess or dispose of it. The majority of fuel in storage is in ponds similar to those provided for short-term storage at reactor or reprocessing sites. With proper control of water chemistry, Zircaloy fuel from light water reactors is expected to retain its integrity for many decades. The main disadvantage of storing large inventories of fuel in cooling ponds is that the ponds require active systems to provide monitoring, cooling, water treatment and ventilation.

An alternative favoured in some countries is storage in dry casks, which are similar to transport containers. These provide radiation shielding and high-integrity containment and can contain up to about 20 fuel assemblies. The heat is dissipated by conduction and natural convection, which offers the benefit of passive safety, that is their safety is not dependent on external power supplies or other services. The casks are stored on a concrete hard standing either in the open air or, more generally, within a simple industrial-style building to give weather protection. They are inspected and monitored according to a defined schedule but require very little maintenance.

Under normal operating conditions, none of the long-term fuel-storage concepts presents any significant radiological problems and operator doses are very low.

Waste and decommissioning

The remaining back-end activities are the eventual decommissioning of nuclear facilities and the management of radioactive wastes. These are addressed in the following chapter.

SUMMARY OF KEY POINTS

Fission: splitting of nucleus into two fission fragments which decay to fission products.

Neutron chain reaction: the process by which some of the neutrons released in one fission cause another fission to occur.

Critical mass: the smallest amount of fissile material needed for a sustained chain reaction.

Reactor system: the core consists of fuel elements which contain the fissile material, control rods and a moderator; other important features are the cooling system, which removes heat from the fuel, and the biological shield, which protects the operators.

Refuelling: the process of removing used fuel from the core and replacing it with unirradiated fuel: carried out on- or off-load depending on the reactor type.

Sources of radiation when operating: fission neutrons and γ-rays, fission product decay γ-rays, neutron capture γ-rays and activation product decay γ-rays.

Shut-down sources: fission and activation product decay γ-rays.

Contamination may occur as the result of a coolant leak or maintenance operations.

Containment: the structures and systems provided to prevent the spread of fission products: vital to prevent overexposures.

Fuel storage pond: a large pool of water which provides heat removal and shielding for irradiated fuel: two special hazards associated with fuel ponds; loss of shielding and criticality.

Fuel reprocessing plant: chemical separation of fuel into uranium, plutonium and waste streams. Severe radiological problems associated with the reprocessing operations and handling the product streams.

REVISION QUESTIONS

1. Describe the process of nuclear fission and explain the circumstances under which a fission chain reaction may be achieved.
2. Draw a simple sketch of a typical nuclear reactor and label the major features of the system.
3. Using a simple diagram, illustrate the sources of radiation from a nuclear reactor system, indicating which are important when:
   (a) operating; and
   (b) shut down.
4. What is meant by containment in the context of nuclear reactors?
5. Discuss the main hazards presented by fuel storage ponds and how they are controlled.
6. Describe the major operations that take place in the 'front end' of the nuclear fuel cycle and discuss the radiological hazards associated with each of them.
7. Describe the major features of a fuel reprocessing plant and the main radiological hazards encountered.

12 Radioactive waste and the decommissioning of radioactive facilities

12.1 INTRODUCTION

Before the discovery of nuclear fission and its utilization as a source of energy, the disposal of radioactive waste did not present a significant problem. It has been estimated that the total quantity of radioactivity in use in research and medicine in 1938 was less than 30 TBq (terabecquerel), corresponding to about 1 kg of radium derived from natural sources. Today, a single large power-generating reactor may contain in excess of 10^8 TBq of fission products, and there are more than 400 power reactors in the world. With the increasing emphasis on protection of the environment, the management of waste is becoming an important factor in both the economics and the public acceptability of nuclear power.

On a much smaller scale, radioactive waste arises in hospitals, factories, research facilities and teaching institutions as the result of a wide range of applications of radiation. In such cases, the complex treatment plants used at nuclear power stations would be prohibitively expensive, and so simpler disposal methods are used which might even be via the normal refuse collection or sewage systems. Clearly, the consequences of all such practices must be understood and strict control exercised.

Radioactivity cannot be destroyed. It will decay eventually but, in view of the very long half-lives of many radionuclides, it is not always practicable to await the decay of radioactive material. There are three general approaches to the management of radioactive wastes:

1. **release** and **dispersal** into the environment;
2. **storage**; and
3. **disposal**.

Of course, release of radioactivity into the environment might reasonably be thought to constitute disposal. However, it is useful to distinguish between deliberate dispersal and methods of disposal involving the irretrievable placement of wastes so that they are isolated, at least temporarily, from the environment.

Storage of radioactive waste is a particularly useful procedure when dealing with nuclides of a relatively short half-life (e.g. up to a few months). Storage or hold-up of the waste for a period of up to a few years may reduce the activity to a sufficiently low level to permit release to the environment or, in the case of solid waste, to facilitate disposal.

The approach selected in a given situation depends on many factors, such as the quantity, the radionuclides involved, their physical and chemical forms, and the geographical location. In this chapter, after a discussion of the consequences of release of

radioactivity, the application of these alternative approaches to the management of liquid, gaseous and solid wastes is discussed.

The structures and equipment contained in facilities generating, handling or processing radioactive materials eventually become radioactive wastes when the facility reaches the end of its useful life and is **decommissioned**. The operations involved in decommissioning radioactive facilities can present significant and novel issues in terms of radiation protection. These issues are summarized in section 12.6.

12.2 CONSEQUENCES OF RELEASES OF RADIOACTIVITY

Any release of radioactive materials into the environment is a potential source of radiation exposure to the population at large. The radiation exposure can occur via many different **exposure pathways**. Consider, for example, a release of activity from a chimney stack. This would be dispersed by air movements and could result in radiation exposure of the population in a number of ways:

1. direct external beta (β) or gamma (γ) radiation from the plume;
2. inhalation of radioactive materials resulting in internal dose;
3. direct external β or γ radiation from deposition (fall-out) of radioactivity;
4. consumption of foodstuffs (e.g. vegetables) contaminated by deposition; and
5. consumption of meat or milk from animals which have grazed on contaminated ground.

Similarly, radioactivity discharged into a river, lake or the sea could result in human exposure via a number of pathways, such as:

1. contamination of drinking water supplies;
2. external dose to swimmers or to people using contaminated beaches; and
3. consumption of contaminated fish, shellfish or plants.

Some of these exposure pathways result from complex routes known as **food chains**. For example, activity discharged to the sea may be taken up by plankton, which are eaten by fish, which are in turn eaten by man. An example of a simple marine food chain is illustrated in Figure 12.1.

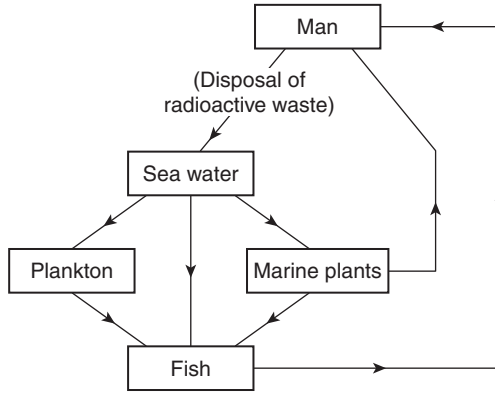


Figure 12.1 A simple marine food chain.

In a given situation, it is often found that one particular pathway is much more important than any other, that is, it results in a much higher radiation dose, sometimes to quite a small group of people. This is called the **critical exposure pathway**, and the group of people that receives the highest dose from this pathway is known as the **critical group**. The importance of identifying the critical exposure pathway is that it allows the quantity of radioactivity discharged to be controlled so as to limit the dose to the critical group. The critical group in any particular case depends on the mode of release, the nuclides involved, local ecology (e.g. forms of marine life, etc.) and local habits.

The fuel reprocessing plant at Sellafield in the UK provides a classic example of the application of this concept. In the plant, irradiated fuel from reactors is processed chemically to recover uranium and plutonium. The other elements, including the fission products and higher actinides, which are intensely radioactive, are transferred to special storage facilities (see section 12.5). However, a small proportion of the activity appears in subsidiary waste streams, which, after a period of storage to permit some decay, are discharged into the Irish Sea.

Before any activity was released from the plant, a detailed study was made to estimate how much activity could safely be discharged. This involved investigations of the local tidal currents and the dispersion that would occur, and investigations of the local ecology. This pre-operational study suggested that there would be three main exposure pathways: deposition of activity on sand and silt, uptake in fish and uptake of activity in a variety of edible seaweed. It was anticipated that the last pathway would be the most important, and this was subsequently confirmed when the plant became fully operational. This pathway arises because one of the fission products, ruthenium-106 (^{106}Ru), is taken up very efficiently and is effectively reconcentrated by the seaweed *Porphyra umbilicalis*. At that time, this seaweed was harvested on the coast close to Sellafield and was used to produce a foodstuff called **laverbread**, which is consumed in South Wales. Thus the factor that controlled discharges from the plant was the dose from ^{106}Ru to the gastrointestinal tract of laverbread consumers. Later, harvesting of *Porphyra* ceased and the emphasis shifted to the other pathways, including plutonium and americium in fish and shellfish.

The significance of deposition of activity on sand and silt is that beaches and estuaries become contaminated and this leads to the external radiation exposure of people using the beaches and of fishermen, for example when handling nets and equipment. Resuspension of activity into the atmosphere when the beaches dry out has been identified as a possible exposure pathway, as was transfer from the sea via sea spray into the atmosphere and then onto land. This could be of particular importance in the case of relatively small quantities of plutonium and other actinides.

The activity taken up by fish is primarily caesium-137 (^{137}Cs). Because caesium is soluble in sea water and because of its 30-year half-life, it disperses over a very wide area, giving a low but measurable concentration not only in the Irish Sea but also in the North Sea and even in the Norwegian Sea. This leads to low but measurable concentrations of the activity in fish caught over a very wide area. One important point that emerged from the experience at Sellafield is that when long-lived activity such as plutonium is released into the environment, its dispersal can be quite slow, thus giving rise to a continuing radiological impact.

Recently, the levels of discharge have been considerably reduced and the radiological impact is low for all of the exposure pathways.

In the case of discharges to atmosphere, an important exposure route is the uptake of radioiodine, mainly iodine-131 (^{131}I), in the thyroid as a result of releases of fission

products to the atmosphere. The uptake may be from inhalation of airborne iodine or consumption of milk from cows that have grazed on contaminated pasture. In either case, exposed children constitute the critical group because of the relatively large intake by children in proportion to the size of the thyroid.

In all of these cases, the discharge limits are set to be as low as reasonably achievable (ALARA) so that the critical group receives doses well below the relevant dose limit.

12.3 RADIOACTIVE LIQUID WASTE

In a typical nuclear facility, substantial volumes of **radioactive effluents** arise from a variety of sources, which could include let-down or leakage from plant systems and processes, cooling pond water, decontamination and laundry wastes, and drainage from active areas. Both the levels of radioactivity and the chemical composition of the effluent arising from these different sources can vary widely, necessitating a range of treatment options which may include the following, often in combination:

- decay storage (for short-lived radioactivity);
- filtration;
- ion-exchange; and/or
- other chemical processes.

The purpose of processing by these means is to reduce the level of radioactivity in the effluent to a level acceptable for release into the environment, usually into a river or coastal waters. After treatment, the liquid wastes are usually collected in one of a number of hold tanks. When a tank is full, it is isolated and mixed thoroughly before sampling and analysis of the contents. There is then a formal process of approval for the discharge of the waste. The hold-up and discharge systems incorporate engineered safeguards, such as interlock and key mechanisms, to minimize the possibility of incorrect operation.

Most treatments lead to the production of other wastes in the form of sludges, resins and filter media, which themselves will require treatment to convert them into a form suitable for disposal as solid waste. Waste treatment involves the provision of additional equipment, which not only involves extra cost but also can result in increased radiation exposure to operators and maintenance workers.

In medical facilities, radioactive effluents arise mainly from the use of radioactive pharmaceutical products in the diagnosis and treatment of disease (see section 14.5). Much of the effluent is biologically contaminated in that it arises as excreta from patients. Here there are great advantages in being able to discharge the effluent into the sewage system, either directly or after a period of decay. Discharge to sewers is also widely used for liquid wastes from research institutions and other facilities where both the volume and the specific activity are low. However, this needs to be strictly controlled. The main considerations are that sewage, sewer walls and sewage works become contaminated, potentially resulting in the exposure of sewage workers. In addition, sewage sludge is often used as an agricultural fertilizer and, if it is contaminated, could result in contamination of crops.

Discharge of radioactivity into rivers is limited by the subsequent use of the river water. Most major rivers are sources of drinking water and of water for irrigation of crops and for the watering of cattle. These considerations mean that discharges into most rivers are limited to quantities of the order of a terabecquerel per year, the actual quantity depending

on local conditions. Similarly, in the case of lakes, the restricted dispersion usually limits the allowable discharge rates of most nuclides to relatively low levels, again typically a few terabecquerels per year.

In principle, relatively large quantities of radioactivity can be discharged into the sea provided that steps are taken to ensure adequate dilution and dispersal. The limits are usually dictated by food chains involving reconcentration effects in marine organisms, as described in section 12.2. At most coastal locations, discharges of some hundreds of terabecquerels would be possible without leading to an excessive dose to the critical group. However, discharge of radioactivity into the environment is a very contentious issue and the emphasis is on ensuring that discharges, and hence critical group doses, are ALARA. The Convention for the Protection of the Marine Environment of the North-east Atlantic (known as the OSPAR Convention), to which the European Union (EU) is a signatory, has set the objective of preventing pollution of the maritime area from ionizing radiation through progressive and substantial reductions of discharges, emissions and losses of radioactive substances, with the ultimate aim of concentrations in the environment being near background values for naturally occurring radioactive substances and being close to zero for artificial radioactive substances. In effect, this removes discharge into the sea as a significant waste management option.

Whatever the mode of liquid discharge, the permitted levels need to be kept under continuous review and the safety of the procedure must be confirmed by environmental monitoring programmes. In the case of marine discharges, this should include radioactivity measurements on seawater, fish, shellfish, seaweed, and the seashore and seabed, as well as radiation dose rate measurements at coastal locations.

12.4 RADIOACTIVE GASEOUS WASTE

Gaseous wastes (within which we usually include streams containing aerosols and vapours) typically arise from the ventilation of radioactive or potentially radioactive areas and from process equipment. The ventilation air generally represents much the greatest volume and the usual method of treatment is high-efficiency filtration and discharge to atmosphere. The process wastes can include extracts from glove boxes and process cells, as well as direct venting from vessels and other equipment. Here the treatment is entirely dependent on the nature of the effluent and therefore on the processes being undertaken. In extreme cases, it could include complex treatments such as electrostatic precipitators or chemical scrubbing systems. As with liquid waste streams, the objective of the treatment is to reduce the concentrations of radioactivity to a level such that the effluent can be discharged into the atmosphere. In some types of installation, it is also necessary to take account of other factors such as the possible presence of toxic or flammable contaminants.

Releases of gaseous or airborne particulate radioactivity to the atmosphere present a more direct exposure pathway than other forms of disposal and, with a few exceptions such as the noble gases, the discharge limits are quite low. The exposure pathways include external irradiation, inhalation and ingestion by various routes (see section 12.2). The general philosophy is to reduce the activity being released into the atmosphere as far as is practicable and then to release it in such a way as to obtain adequate dispersal.

When relatively small amounts of activity are involved, the releases to the atmosphere are usually from an extract discharging at, or even below, roof level. Care must be taken

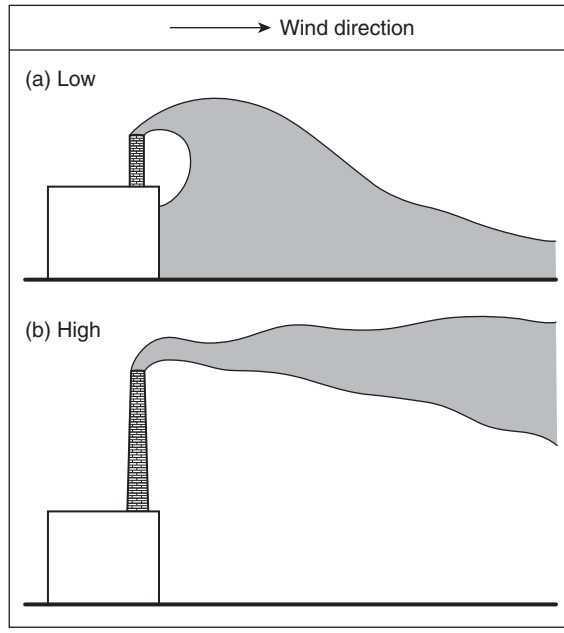


Figure 12.2 Dispersion from chimney stacks.

in the siting of such extracts since, under certain weather conditions, eddies and currents may cause the released activity to re-enter the building through air intakes or even through open windows. A chimney stack is preferable but the additional cost is only justifiable when large quantities of activity are being released. Ideally, the chimney should be two to three times the height of surrounding buildings to obtain good dispersal. This is illustrated in Figure 12.2, which shows the dispersion from (a) a short stack and (b) a tall stack on a large building. Even with the tall stack, diffusion or spreading of the plume means that, at some distance downwind, the concentration of activity at ground level will show an increase. The maximum concentration usually occurs at a distance downwind of 10–20 times the height of the stack, depending on wind and weather.

As with liquid wastes, the adequacy of controls is confirmed by the sampling and measurement of discharges and by detailed environmental monitoring programmes.

12.5 RADIOACTIVE SOLID WASTE

Radioactive solid wastes arise in various forms in nuclear facilities and from medical and industrial applications of radioactivity. They are usually considered to fall into three broad classes: low-, intermediate- and high-level wastes.

Low-level waste (LLW) consists typically of general trash from contaminated areas as well as items of lightly contaminated or activated plant and equipment. **Intermediate-level wastes** (ILW) arise mainly in nuclear facilities and include solidified process residues and significantly activated items. The definitions of LLW and ILW, in terms of specific activity, vary from country to country but, typically, LLW would comprise waste of less than about 10^9 Bq/m^3 and ILW would be any waste other than high-level waste (HLW), as

defined below, exceeding this value. With both LLW and ILW, disposal is often facilitated by segregation of the waste, preferably at the point of arising, for example into wastes of different specific activity, or into combustible and non-combustible material. With the high cost of waste disposal, there is strong incentive to minimize arisings of waste as far as possible. For example, on a practical level, LLW arisings can be reduced by avoiding taking unnecessary items and materials into contaminated areas since these can then become contaminated and thus constitute radioactive waste. Another method of reducing arisings of some classes of waste is to recycle the materials. This has been applied particularly to activated steel from dismantled reactors, which can be recycled through a steel foundry where it is diluted with non-active steel and used for components for new nuclear facilities.

A reduction in volume is usually beneficial, and this can be achieved either by compaction of the waste or, in the case of combustible waste, by incineration. In the latter case, the fuel gases from the incinerator are normally filtered and the filters and ash then constitute the reduced volume of waste. In the UK, a further category, **very low-level waste** (**VLLW**), is recognized. This is material that is only mildly radioactive and that can in many cases be disposed of by conventional landfill.

The term **high-level waste** (**HLW**) is usually reserved for the raffinate from the first extraction cycle in a fuel-reprocessing plant. This stream contains 99 per cent or more of the fission product activity from spent nuclear fuel and much of the α-emitting higher actinides. The specific activity of this waste is so high that it generates significant heat, and special provision has to be made for cooling. Initially, the waste is stored as a liquid in high-integrity tanks with multiple cooling systems and located inside massive concrete-shielded cells. After a delay of a few years to allow some of the shorter-lived activity to decay, the waste is vitrified, which means that it is incorporated into large glass blocks in stainless steel canisters. The waste is then in a suitable form for safe storage for an extended period.

Storage can only be regarded as a temporary solution to the problem of managing radioactive waste. This is because many types of waste contain long-lived radioactivity. For example, the activity in HLW includes the fission products strontium-90 and caesium-137, with half-lives of 28 and 30 years, respectively. After about 500 years these will have decayed to negligible levels, but significant amounts of longer-lived nuclides will remain, including the α-emitters americium-241 (^{241}Am, half-life 434 years), plutonium-239 (^{239}Pu, half-life 24 000 years) and neptunium-237 (^{237}Np, half-life 2.2 million years).

The need to establish safe methods of radioactive waste disposal is an important factor in the public acceptance of nuclear power. In some countries, further development of nuclear power has been made dependent on establishing the feasibility of the safe disposal of wastes.

A method practised by smaller users of radioactive materials, such as hospitals, universities and industrial users, is disposal with ordinary trade waste. The principle being applied here is **dilution**, the odd package containing a small quantity of radioactivity being well diluted and buried among the vast quantities of ordinary trade and domestic waste on the local refuse tip. This approach requires very strict control and the permitted quantity per package depends on the radionuclide composition.

For the much larger volumes and higher radioactive concentrations of wastes arising from the nuclear industry, two types of special disposal facilities, or waste repositories as they are known, are in use or planned. The first is a **shallow land facility** in the form of a large engineered concrete structure 10–30 m below the surface, usually in clay beds. This type of repository is suitable mainly for LLW, but may also be suitable for short-lived ILW.

The second type is a deep **underground** disposal facility, perhaps about 300 m or more below the surface. During the emplacement phase, such facilities would be similar to mine workings, with shafts and tunnels. The wastes, contained in high-integrity packages, would be placed in tunnels which would then be backfilled with absorbent materials and, eventually, the workings and shafts would be completely sealed. This type of repository is potentially suitable for both ILW and HLW.

The important consideration in underground disposal is that the waste containment will eventually be lost and activity will be leached into groundwater, although this could take thousands or even hundreds of thousands of years. This could result in the contamination of drinking water supplies or of crops. Disposal sites must be carefully selected to minimize these effects. Another consideration, particularly for shallow land sites, is the possibility of inadvertent human intrusion at some future time. Examples of intrusion are drilling of boreholes, tunnelling and excavation. While controls over the development of the site should be able to prevent these situations for some hundreds of years, beyond this time it is not possible to guarantee that intrusion would not occur. For this reason, the amount of long-lived waste placed in shallow land repositories will need to be strictly limited.

12.6 DECOMMISSIONING OF RADIOACTIVE FACILITIES

With many of the first-generation nuclear research and power-generation facilities now closed down, the decommissioning of redundant nuclear plants has become a major issue in the safety, economics and environmental impact of the nuclear industry. The early plants were constructed without any consideration of the eventual need for their decommissioning and, in many cases, this has necessitated complex and expensive approaches. In later plants, the need to make suitable design provision to facilitate decommissioning was recognized, and this is expected to yield both radiological and economic benefits in the long term.

From a radiological point of view, the aim of a decommissioning programme is to achieve a progressive reduction of the hazard posed by the plant, while ensuring that the health and safety of the decommissioning workers is protected and the impact on the general public and the environment is minimized. The major differences between different types of nuclear facilities, in terms of size, complexity and radioactive inventory, mean that the specific approaches and details of the decommissioning programmes for different nuclear facilities vary considerably, as discussed in the sections below.

The special problems that arise in dealing with redundant nuclear facilities come from the radioactivity remaining in the plant at the end of its useful life. This has three technical implications:

- A high standard of containment of the radioactivity is needed in order to protect the local population.
- The residual radioactivity in the plant poses a radiological hazard to workers involved in the decommissioning of the plant.
- The radioactive structures and equipment are important sources of radioactive waste streams, both directly and because of the secondary wastes that arise from decontamination and dismantling operations.

These all need to be taken fully into consideration in the planning and preparation activities, which are a vital part of any decommissioning project.

12.6.1 Planning and preparation

The key to a successful decommissioning project is pre-planning, and this normally commences some years before the scheduled closure. The first steps are to prepare a detailed **inventory** of the plant, equipment and structures and a full radiological characterization. The characterization is based on a detailed programme of measurements supplemented, in most cases, by calculations. This allows the radioactive inventory of the many different components of the plant to be estimated as well as the probable waste arisings and the types of waste packages likely to be needed. It also permits estimates to be made of the radiological hazards and risks associated with dismantling operations in order to provide a basis for selection of the most suitable engineering approaches. For example, where the assessment shows that very high radiation levels are likely to be encountered, consideration would need to be given to the use of remotely operated equipment. The radiological characterization is also a vital first step in the production of an ALARA strategy for minimizing the radiological impact of the decommissioning project. This ALARA strategy needs to take account of all relevant factors such as the sources of radioactivity and radiation on the plant, the working conditions and the proposed engineering approach. The strategy needs to be confirmed by actual measurements and experience and should be flexible enough to cater for changing circumstances during the course of the project.

12.6.2 Stages of decommissioning

As noted earlier, the aim of a decommissioning programme is the progressive reduction of the hazard posed by the plant. For this reason, the first stage of decommissioning is a general **clean-up of the facility** and the removal of process materials and wastes. The nature of the tasks involved is often similar to those encountered during operation. In the case of a reactor, the major activity at this stage is the **defuelling of the core** and the removal of fuel from the site. This removes a large proportion of the radioactive inventory and essentially eliminates the possibility of a major release. For process plants, the first stage is normally a post-operational clean-out of process equipment, often referred to as POCO, to remove as much of the process material as possible.

The second stage normally involves the **removal of the auxiliary plant and equipment**, particularly those items that are not radioactive and are not needed to support subsequent decommissioning operations.

The final stage is the **dismantling** of all the remaining facilities, removal of the waste and the release of the site either for re-use as a nuclear site or for an alternative use. However, there are various alternative approaches that may be adopted, varying from immediate dismantling to deferred dismantling after an extended period of protected storage or a combination of the two. The approach adopted for any particular facility depends on a number of factors, including the radioactive inventory and its pattern of decay and the relative costs of different options. There are important differences of principle involved in the decommissioning of reactors compared with fuel cycle facilities or other radioactive plants. In particular, the residual activity in a reactor is almost entirely in the form of activated structural materials, while in other plants the radioactivity is in the form of process residues and general contamination of plant structures and equipment. Thus decontamination of a reactor system, although often a useful step, might not significantly reduce the overall radioactive inventory or the radiological impact of dismantling the plant. In other types of plant, early decontamination is a key factor in relation to the exposure

of personnel and the management of radioactive waste. Another difference is the pattern of decay of the radioactivity. In the case of a reactor, the activation product cobalt-60 (half-life 5.2 years) usually dominates the inventory, and so substantial benefits, in the form of reduced operator exposure and lower waste arisings, can be gained by delaying the dismantling for a few decades. In fuel plants, little advantage accrues from such a delay because of the long half-lives of many of the radionuclides present in the plant.

This staged approach is illustrated below by reference to the decommissioning of nuclear power plants, research reactors and other plants in the nuclear fuel cycle.

Nuclear power plants

In the case of a large nuclear power plant, stage 1 decommissioning normally includes:

- defueling of the reactor and, if possible, removal of the fuel from the site after a period of decay storage;
- decontamination and general clean-up of the ancillary facilities; and
- where a disposal route is available, packaging and removal of accumulated radioactive wastes.

In stage 2, all equipment and buildings that can be easily dismantled and are not needed for future decommissioning operations are removed or are decontaminated and made available for other uses. If the final dismantling is to be deferred for an extended period, the containment and/or the biological shield structure may be extended and sealed to enclose the reactor structure. The aim is to protect people and the environment from the residual radioactivity but also to protect the plant and minimize its deterioration over the extended period of storage. Over the storage period, a routine programme of maintenance and surveillance of the containment is carried out as well as radiation and contamination monitoring of the environment.

The third stage is the dismantling of the reactor structure and all other redundant facilities on the site. This is the most difficult, expensive and hazardous part of the whole process since it involves the removal and cutting up of very large radioactive structures and components of steel, concrete and other materials. Although a period of protected storage will have resulted in some reduction of the radioactive inventory, and hence of the radiation hazards, there will still be the need to maintain high standards of shielding and containment to protect workers and the off-site population. This usually involves the construction of further containment structures and of facilities for cutting and handling the radioactive materials as they are removed from the reactor. Also, there is usually the need to re-establish equipment such as ventilation plant and solid and liquid waste treatment facilities. At the end of the reactor dismantling phase, these supporting facilities will themselves need to be dismantled.

Research reactors

A considerable number of research reactors have already been decommissioned worldwide which have generated a large amount of relevant experience on the radiological, environmental and waste management challenges involved in such activities. The three-stage approach described for power reactor decommissioning is equally applicable to research reactor decommissioning. However, research reactors have a number of special features that need to be considered when developing the most appropriate ALARA strategy for decommissioning, including:

- Their radioactive inventories, of both fission and activation products, are generally much smaller than those of power reactors.
- Most research reactors use special, non-commercial fuel which may be highly enriched and have rather exotic cladding materials.
- The special nature of the core structural materials, control rods and coolants of some research reactors may introduce unique handling issues.
- The research activities carried out on any particular reactor, such as the use of fast neutron beams for irradiation, may have left some specific legacy issues.

The siting of some research reactors may introduce special problems in relation to estimating and controlling the individual and collective doses to members of the public and ensuring that the local environment is adequately protected.

Other facilities in the nuclear fuel cycle

As discussed in section 11.8, the complete nuclear fuel cycle includes 'front-end' facilities for the production of fuel and 'back-end' facilities for the management of spent fuel and the associated products and waste streams.

The clean-up and decommissioning of **front-end facilities** present relatively few radiological control issues. The main issue during the decommissioning of enrichment plants is handling and disposing of the vast quantities of **depleted uranium** (**DU**), which are often stored in or around such plants. DU presents a problem because of the chemical toxicity of uranium hexafluoride (UF_6). The decommissioning of most fuel fabrication plants presents few radiological challenges apart from the need to ensure suitable protection of the workers from the UO_2 dust, which remains in the plant as the result of grinding operations. The only exception is where the plant has been used to fabricate mixed uranium–plutonium (MOX) fuel. In such situations, the radiological challenges faced during decommissioning can be quite severe because of the high radiotoxicity of plutonium compared with uranium.

Decommissioning the plants at the **back end of the nuclear fuel cycle**, however, often presents major radiological control challenges. The fuel storage ponds may contain damaged fuel elements, significant quantities of radioactive sludge (from the corrosion of fuel cladding and other components) and other activated metal components as well as the cooling water, which is itself often contaminated. If the ponds are old, there may be questions about their structural integrity and their ability to support the machinery needed for clean-out and decommissioning. Fuel reprocessing plants present special hazards associated with the chemicals used in the reprocessing process and the very long-lived radioactive elements that are extracted during the reprocessing process. These include the fission products such as ^{137}Cs and ^{90}Sr and very long-lived α emitters such as plutonium and americium. The clean-out and decommissioning of such plants involves the extensive use of remote handling equipment and special protective clothing in order to minimize the dose (both external and internal) to the workers. The steps that have to be taken to assess the risk of such decommissioning activities are illustrated in Example 2 in section 15.4.

12.6.3 Radiological protection in decommissioning

The health and safety of workers and the protection of the general public and the environment are key factors in the selection of strategies and approaches to

decommissioning. To this end, a radiological impact assessment should be made to determine:

- individual and collective doses to workers during the clean-up and dismantling processes, including those from waste handling;
- individual and collective doses to the public throughout the entire operation, including those resulting from waste disposal and any activity remaining on the site;
- the potential risks of an uncontrolled release of radionuclides to the environment as a consequence of possible deterioration of the installation or the possible loss of integrity of its containment barriers. It needs to include an assessment of other potential hazards such as fire, explosion, and industrial and chemical hazards.

The nature of decommissioning operations is such as to require a high level of health physics surveillance. During dismantling of equipment and structures, there is always the potential for an unexpected radiological hazard as the consequence, for example, of a release of contamination or the loss of shielding of a radiation source. Regardless of how well the plant has been characterized in advance, events such as this can occur, and this means that radiological conditions need to be monitored continuously during dismantling operations. Levels of worker exposure also need to be monitored continuously using electronic dosimeters with direct readout. The standards of protective clothing and equipment worn by workers need to be kept under review. Certain operations might require full respiratory protection, but if excessively applied, the result could be to slow down operations and increase the dose from external radiation. All decommissioning projects should be carried out within a properly defined management regime using a predefined and approved methodology, but it is important that the management and approval regime is sufficiently flexible to be able to take advantage of the lessons learned as the project proceeds.

12.6.4 Site release

At the end of decommissioning, it is important to decide whether the land on which the nuclear facility was sited can be released. Even after the plant and structures have been removed, there is always the possibility that low levels of radioactivity will remain on or under the surface of the site. Before such a site can be released for other use, it is necessary to go through a formal monitoring and clearance process. The monitoring involves a detailed programme of both direct radiation monitoring and soil sampling for laboratory analysis. When significant levels of site contamination are detected, a programme of **site remediation** must be undertaken. This might entail the excavation of areas of the site and removal of the soil to a suitable approved landfill site. In other cases, depending on the levels and extent of the contamination, the radionuclides present and the planned use of the site, it may be acceptable to bury the contaminated soil directly on the site, ensuring that there is an adequate thickness of cover. The criteria for clearance of sites vary to some extent from country to country. For clearance of nuclear sites in the UK, the regulatory authority considers that any residual radioactivity above the natural background that can be demonstrated to pose a risk to any person of less than one in a million per year would be broadly acceptable. This corresponds to an average radiation dose rate above that from natural background of about 20 μSv/year.

12.7 REGULATIONS

The legislation and regulations covering the management and disposal of radioactive waste vary from country to country. Within the EU, member states implement the Basic Safety Standards Directive in national legislation. For example, in the UK, those aspects of the Directive relating to radioactive waste are implemented mainly through the provisions of the Environmental Permitting Regulations (2010) in England and Wales and the Radioactive Substances Act (1993) in Scotland and Northern Ireland.

The majority of small users of radioactive sources are required to register the use and storage of radioactivity with the Environment Agency under the provisions of the Regulations. Authorizations for disposal are granted by the Agency after consultation with others, including local authorities.

Operators of nuclear installations such as nuclear power stations and fuel manufacturing and reprocessing plants are exempted from the requirement to register their use and storage of radioactive materials. Instead, they are subject to the provisions of the Nuclear Installations Act (1965, 1969) and amendments. In this Act, reference is made to the Environmental Permitting Regulations and the responsibility for authorizing discharges again rests with the Environment Agency. In Scotland, the responsibility for authorizing discharges rests with the Scottish Environment Protection Agency.

Within the EU member states, nuclear reactor decommissioning projects are required to adhere to a framework set out in Council Directives 85/337, 97/11 and 2003/35, which require an environmental impact assessment (EIA) to be carried out for certain types of projects. With regard to nuclear reactor decommissioning, these requirements are implemented in the UK through the Nuclear Reactors (Environmental Impact Assessment for Decommissioning) (Amendment) Regulations 2006 (EIADR06). The Regulations require that prior consent is obtained before the commencement of any decommissioning operations and this consent would be based, among other things, on the submission and acceptance of an EIA. Following consent, the decommissioning operations are undertaken within a similar regulatory regime to that which applies during facility operation.

12.8 TRANSPORT OF RADIOACTIVE MATERIAL

Transport of radioactive material by land, sea and air has been practised on a large scale for over 60 years and it is estimated that, worldwide, some 30 million shipments take place annually. These include the carriage of radioisotopes for industrial, medical and research applications as well as shipments of fuel cycle materials and radioactive wastes by the nuclear industry. It was appreciated at an early stage that, in order that consignments could cross national borders, there was a need for internationally recognized standards. The first comprehensive regulations were those issued by the International Atomic Energy Agency (IAEA) in 1961. The fundamental principles embodied in the regulations have proved to be sound and remain essentially unchanged, although their detailed application has evolved in a series of revisions, (IAEA, 2009). As with most other international regulations, the transport regulations are given force by being incorporated into national legal frameworks.

Radiation protection of workers and members of the public from transported material is achieved by applying the general principles of containment of the radioactive content, control of external radiation levels, prevention of criticality (where fissile materials are

involved) and prevention of damage caused by heat. The nature of radioactive materials shipments vary widely from small sealed sources or small amounts of radiopharmaceutical preparations in a cardboard box up to intensely radioactive spent nuclear fuel in steel containers weighing 60 tonnes or more. The regulations adopt a graded approach to package contents limits and performance standards for package designs. These performance standards for packages (defined as comprising the radioactive material and its packaging material) need to take account of the conditions to which a package might be exposed during its transport, including routine incident-free conditions, normal conditions, including minor mishaps such as dropping or getting wet, and severe accident conditions. The regulations also specify requirements for labelling and for limits on the radiation levels in the vicinity of packages.

SUMMARY OF KEY POINTS

Main sources of radioactive waste are the nuclear fuel cycle and the use of radioisotopes in medicine, industry and research.

Principles applied are release and dispersal, storage and disposal.

Consequences of disposal: dose to population via exposure pathways, direct or via food chains. The emphasis is on ensuring that the doses received from waste disposal operations are ALARA.

Limiting pathway is the critical exposure pathway, and the population group receiving highest dose is the critical group.

Liquid waste: low-level liquid wastes discharged to environment. High specific-activity wastes treated by ion exchange, evaporation or chemical treatment.

Gaseous waste: release to atmosphere after filtration or scrubbing. Discharge from stack to achieve good dispersal.

Solid waste: LLW, ILW and HLW. Storage is only a temporary solution for long-lived wastes. Segregation and volume reduction facilitate disposal. Possible routes are shallow or deep underground disposal.

Regulations: authorization is required for storage or disposal of radioactive waste.

Decommissioning: aim is to achieve a progressive reduction in the hazard

Decommissioning strategy depends on facility but usually is undertaken in three stages, with dismantling as the final stage.

Planning and preparation are the keys to a successful programme and should include a detailed ALARA strategy.

Site release requires a formal monitoring and clearance process. In the UK, the requirement is to demonstrate that the residual radioactivity poses a risk of less than one in a million per year.

Transport of radioactive material is based on common international standards.

Transport safety is achieved by containment, limitation of radiation levels, control of criticality risks (where appropriate) and prevention of heat damage.

Package designs and performance standards are based on a graded system which recognizes the very wide range of radioactive materials involved.

REVISION QUESTIONS

1. List the three general approaches for dealing with radioactive waste. Give a practical example in each case.
2. Discuss the concept of critical exposure pathway and give an example of such a pathway that involves a food chain.
3. What are the alternative discharge routes for low-level liquid waste? Discuss the factors which limit the quantities that may be discharged in each case.
4. What are the possible exposure pathways resulting from releases of radioactivity to the atmosphere? How would the exposure from these pathways be limited?
5. Discuss the possible disposal routes for solid radioactive waste. Broadly, what types of wastes are suitable for each route?
6. Discuss the scope of the three stages of decommissioning.
7. What factors could lead to deferral of some of the stages? Illustrate these factors by reference to the decommissioning of (a) a nuclear reactor and (b) a nuclear fuel-reprocessing plant.

13 Radiation protection in the non-nuclear industry

13.1 INTRODUCTION

As well as in the nuclear industry, artificial sources of ionizing radiation are used extensively in a range of other industries. Examples of such use include: X-ray machines and sealed sources for industrial radiography (non-destructive testing) and for other industrial inspection purposes; sealed sources as gauges or measuring devices in the paper industry and in construction; and unsealed sources as radioactive tracers. Research and development facilities make use of a variety of radiation sources as do teaching institutions such as universities, colleges and schools. In order to carry out any of these activities safely an appropriate level of radiation protection is required. Some industrial activities, such as mining, may expose workers to significant amounts of naturally occurring radioactive material (NORM), which necessitates the implementation of radiation protection measures.

The medical and dental sectors are by far the biggest users of radiation machines, especially X-ray machines, and the relevant radiation protection procedures are covered in the next chapter. However, it should be noted that veterinary surgeons make similar use of X-ray machines for diagnostic purposes.

13.2 X-RAYS

13.2.1 General

X-rays were discovered by the German physicist Wilhelm Conrad Roentgen in 1895. Soon after, he found that he could photograph the interior of objects (including human body parts) by passing X-rays through them. Such photographs became known as **radiographs**. The medical implications of this discovery were immediately appreciated and, within a few months, physicians in many parts of the world were using X-rays as an aid to diagnosis. X-rays are now widely used in medicine, and this topic is dealt with in Chapter 14. X-rays also have many other applications in industry and in research.

The most important method of producing X-rays depends on a process known as ***bremsstrahlung***, which is German for **braking radiation**. Bremsstrahlung X-rays are produced when charged particles, usually electrons, moving with a very high velocity are slowed down rapidly by striking a target – for example, when β-particles from a radioactive substance impinge upon a shielding material.

The efficiency of X-ray production by this means is dependent on the atomic number (Z) of the target material, with high-Z materials giving a much higher yield than low-Z materials. (This is the reason for using low-Z materials, such as Perspex, for shielding

beta sources.) In any case the intensity of X-rays produced by β-particles from radioactive substances is too low for most applications. The method used to produce X-rays for medical and industrial purposes is based on an electrical discharge tube and is similar to the method developed by Roentgen, though modern equipment is much safer and more efficient.

Like light, radio waves and γ-rays, X-rays belong to the electromagnetic group of radiations. They have no mass or charge, but have a wavelength which depends on their energy. They differ from γ-rays in two important respects. First, γ-rays originate within atomic nuclei whereas X-rays originate from changes in the electron orbits. Second, γ-rays from a given source have definite discrete energies but X-rays from an X-ray generator usually have a broad range or spectrum of energies up to some characteristic maximum value.

13.2.2 X-ray equipment

X-rays are produced when electrons moving with high velocity are suddenly stopped by a material of high atomic number, and so an X-ray generator requires a source of electrons, a means of accelerating them to a high velocity and a target at which they are directed. An X-ray set consists of a tube and various electrical circuits, which are usually in a separate control unit.

The modern type of X-ray tube, shown diagrammatically in Figure 13.1, consists of a **cathode** and an **anode** inside a glass tube evacuated to an extremely low pressure. The cathode is the source of the electrons and consists of a tungsten filament heated to incandescence by an electric current which 'boils out' electrons. The electrons are accelerated to the target by a high voltage applied between the anode and cathode.

The **target** is part of the anode assembly and is constructed of a material of high atomic number to achieve the best possible efficiency of X-ray production. However, even when the efficiency is as high as practicable, less than 1 per cent of the energy of the electrons appears as X-rays. The remainder appears as heat and so the target must have a high melting point and be able to dissipate the heat. This is achieved by constructing the anode of copper, which has a high thermal conductivity, with a tungsten target insert facing the cathode.

The **copper anode** is sometimes in solid form and has a finned radiator extending outside the tube to assist cooling. In higher power sets the anode is hollow and is cooled

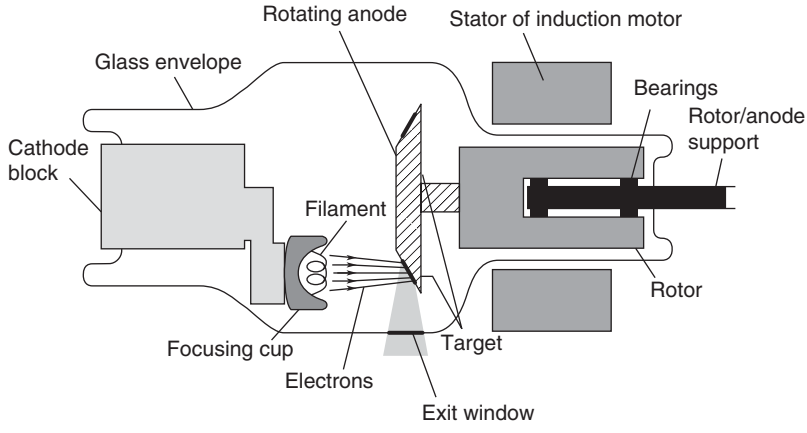
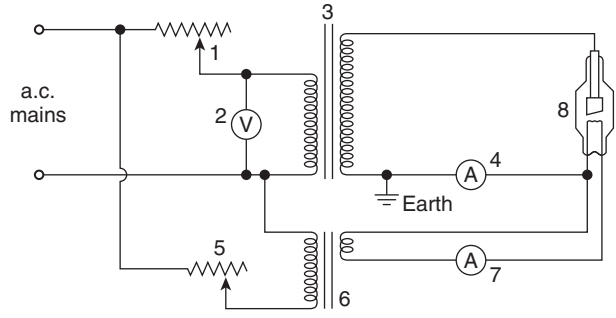


Figure 13.1 A typical rotating anode X-ray tube.

by circulating oil or water through it. In applications such as radiography it is important, in the interests of good definition, that the source of X-rays is very small. The filament is therefore mounted in a concave cup that focuses the electrons onto a small area of the target. Special measures are then necessary to prevent overheating of the target and the anode may consist of a rotating disc. The effective target area is then still small but the heated area is greatly increased and the tube may be heavily loaded without melting the target. This type of tube is used in medical X-ray sets, in which very high intensities and short exposure times are used to minimize difficulties caused by body movement.

The **electrical supplies** required for the operation of an X-ray tube are a low-voltage supply to the filament and a very high voltage supply applied between anode and cathode. The supplies are usually derived from mains alternating current (a.c.). In the case of the filament supply, a step-down transformer is used to provide a voltage of about 12 V a.c. at a current which can be varied up to a few amperes. The tube voltage is provided by a high-voltage transformer which steps up the mains voltage (230 V a.c. in the UK) to the level required for operation of the tube. This is normally in the range from 5000 V up to some millions of volts, depending on the application. As it is derived from a.c. mains, the high voltage between the anode and cathode, if unmodified, would be also alternating, or at least self-rectified to 'half-wave' by the inherent action of the X-ray tube. This is a very inefficient way to generate X-rays. A typical power supply is illustrated in Figure 13.2.



1. Tube voltage control
2. Voltmeter
3. High-voltage transformer
4. Tube current ammeter
5. Filament current control
6. Low-voltage transformer
7. Filament current ammeter
8. X-ray tube

Figure 13.2 Simplified power supply for X-ray set.

Most modern X-ray generators, especially those used for medical applications, are based on a 'high-frequency' or 'constant potential' design. The generator can be thought of as three sub-circuits:

(a) initial rectification and smoothing of the incoming a.c. supply;

(b) a frequency multiplying circuit; and

(c) a high-voltage transformer, rectification and smoothing circuit.

This converts the incoming mains a.c. into a high-voltage waveform which is almost constant with a ripple of less than 1 per cent.

13.2.3 Quality and intensity of X-rays – the X-ray spectrum

The quality or energy of X-rays depends on the voltage waveform applied to the anode of the tube. If the peak voltage is 200 000 V, this is expressed as 200 kV peak or 200 kVp. The maximum energy of the X-rays produced is 200 keV, but only a very small fraction will have this value and most of the X-rays will be of lower energy. The quality of the X-rays is, however, defined in terms of this peak energy and they are said to be 200-kVp X-rays. The penetrating power of X-rays is highly dependent on their energy. For example, the quality of X-rays used to radiograph a person's hand would be much too low to radiograph a 10 mm steel plate. The voltage on the tube is therefore set to give the appropriate quality of X-rays for each application. The spectrum of X-ray photons produced by a typical X-ray tube and generator combination is shown in Figure 13.3. A few examples of suitable operating voltages and exposures for medical and industrial radiographic applications are given in Table 13.1.

While the voltage on the tube controls the quality of the X-rays, the intensity is also governed by the current flowing in the tube, that is, between the anode and cathode. This current, expressed in milliamperes (mA), is limited by the number of electrons ejected from

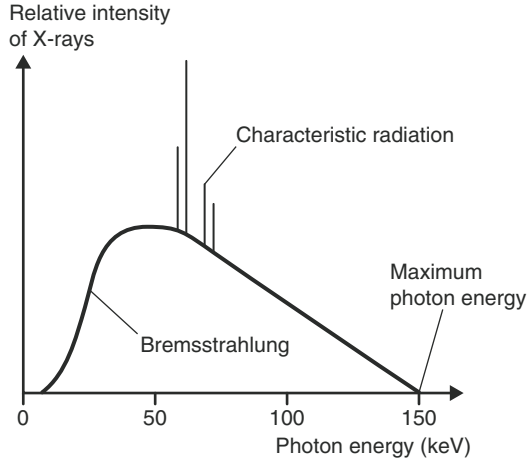


Figure 13.3 The spectrum of X-ray photons from a typical X-ray tube and generator.

Table 13.1 Typical operating voltages for radiography

		kVp	Distance (m)	Milliampere-seconds (mAs)
Medical diagnosis	Dental intra-oral	70	1	10
	Computed tomography abdomen	80	1	100
	Chest	80	2	3
	Pelvis	120	1	30
Industrial	6 mm steel	120	0.5	10
	25 mm steel	160	0.5	200

the cathode. This is controlled by the temperature of the filament, which in turn depends on the current flowing in the low-voltage filament circuit. As the dose rate depends on the tube current, the total dose in a particular case depends on the tube current multiplied by the time of exposure. At a fixed tube voltage, the same dose would be received from an exposure of 10 mA for 1 s as for 1 mA for 10 s. In both cases the exposure is 10 milliampere-seconds or 10 mAs.

The dose rate from an X-ray set is very high compared with dose rates from typical sealed γ sources. The output is usually expressed in terms of the absorbed dose rate in mGy/min to air at 1 m from the set for a tube current of 1 mA. Some typical outputs are shown in Table 13.2.

Table 13.2 Typical output of X-ray sets

Equipment and filtration	mGy/min/mA at 1 m
50 kVp beryllium window tube	100
100 kVp 3 mm aluminium (external to tube)	30
200 kVp 2 mm copper + 1 mm aluminium (external to tube)	20
300 kVp 3 mm copper + 1 mm aluminium (external to tube)	10
500 kVp 3 mm copper + 1 mm aluminium (external to tube)	25

The significance of the beryllium window mentioned in Table 13.2 is that at low voltages the penetrating power of X-rays is so low that a high proportion would be absorbed by a glass bulb. The use of a thin beryllium window minimizes this loss of output.

At higher voltages additional absorbers, or filters as they are known, are provided in the X-ray beam. It has been mentioned that X-rays of all energies up to the peak voltage are produced. Only the small fraction with the higher energies is useful, the remainder being undesirable in many cases. For example, in medical radiography, the low-energy fraction would not contribute to the radiograph, but would result in unnecessary dose to the patient. The use of filters, usually an appropriate thickness of aluminium, selectively absorbs the low-energy or soft radiation without significantly affecting the useful beam.

13.2.4 Protection against X-rays – general principles

Unlike radioisotopes, which emit radiation continuously, X-ray sets can be switched on or off at will. During operation the dose rate from the set may be very much higher than from small sealed sources. The equipment must be run in such a way that the operator does not expose any part of his or her body to the direct beam and no other person should be inadvertently exposed. The general principles applied to the protection of personnel are as follows:

1. adequate training of all personnel who operate or use X-ray equipment in the correct operating procedures and in the hazards involved;
2. limitation of the beam size to the minimum necessary by the provision of shielding and having collimators built into the set;
3. the use of suitable filtration to remove unwanted soft radiation;
4. operation of X-ray equipment in a shielded room whenever possible. The controls are located in a shielded position either inside or outside the room and, depending upon the application, an interlock circuit may be used to prevent

operation of the equipment while the door is open. A notable exception is where X-ray machines are used 'in the field' for mobile radiography;

5. provision of automatic visible and audible warnings that the X-ray set is operating or about to operate;
6. confirmation of the effectiveness of the control measures by means of a system of personal and area monitoring.

The measures applied in any given case depend very much on the type of work and the local conditions. The main applications of X-rays are in industrial and medical radiography. This chapter deals with protection in industrial applications and research, and medical uses are considered in detail in Chapter 14.

13.2.5 Monitoring of X-ray installations

An important part of the commissioning procedure of any radiographic installation, or other facility which produces ionizing radiation, is a thorough radiation survey. Particular attention is paid to possible weaknesses in shielding, such as joints in the shielding material, viewing windows, doors and holes or ducts for services. The survey, which is usually performed at maximum tube voltage and current, is made under normal operating modes and then under other possible operating modes. Consider, for example, the case of a facility in which the X-ray beam is intended to operate in the horizontal plane and adjacent areas are shielded by thick walls. If the orientation of the set is changed and the beam operated in a vertical direction, would unacceptable levels of radiation occur in the areas above or below the facility? It should be borne in mind that if such a change is possible it is quite likely that, one day, it will be made. Even if the areas above or below the cell are unoccupied, high dose rates can occur outside shielding because of radiation scattered from the object being radiographed, the walls, floor or ceiling of the room, or even the air (which is often referred to as skyshine). If it is found that excessive radiation levels could possibly occur in adjacent areas, measures must be taken to prevent, or at least to give warning of, the situation. This can be done by mechanically preventing the beam from being operated outside set directions, by the provision of additional shielding, or by the installation of area radiation monitoring equipment with warning signals. In general, one of the first two methods is preferable.

Clearly, questions of this sort should be considered at the design stage as part of a prior risk assessment, but it is essential to confirm the safety of the facility by direct measurement. Surveys should be repeated periodically, particularly when any changes in operating procedure are introduced.

Care is necessary in the selection of instruments for monitoring X-rays. The major problem is that of energy response. Many instruments that are satisfactory for γ-rays and higher energy X-ray work seriously underestimate the dose rate from X-rays of below about 100 kVp. For low-energy work, instruments incorporating thin-window ionization chambers are probably the most suitable, although they are sometimes lacking in sensitivity. Another possible problem when pulse-type instruments (e.g. Geiger–Müller tubes) are used is that the instrument may saturate in high X-ray dose rates and yet still appear to be working satisfactorily. This is due to the pulsed nature of the X-rays, which allows the instrument to recover between pulses. The instrument might then record the X-ray pulse rate rather than the average dose rate. This problem is fortunately rare in equipment of modern design but may arise as a result of a fault.

The safety of a facility is ultimately judged by the radiation doses received by operators and other persons working in the vicinity. These are normally measured by thermoluminescent dosimeter (TLD) badges, although many establishments still use film-based dosimeters. It is often worthwhile using a few TLD badges or other dosimeters to monitor fixed locations around the area on a routine basis. It should be borne in mind that personal monitors are small in area and X-ray beams, particularly in crystallography, may also be small in cross-section. It is quite possible for a beam to miss a personal monitor but nevertheless to irradiate the worker.

13.2.6 Protection in industrial radiography

The general principles that apply to the control of hazards from industrial radiography are as follows:

1. Non-destructive testing using ionizing radiation should be used only where it offers a clear advantage over other methods, in other words, the use of radiation is justified.
2. Whenever practicable, radiography or any other process using ionizing radiation from machines or sealed sources should be carried out within a shielded enclosure.
3. The control panel for the apparatus should be located outside the enclosure and devices should be provided to ensure that, if any door of the enclosure is opened while the apparatus is energized, the apparatus is automatically de-energized.
4. For the protection of persons accidentally shut inside the enclosure, a means of communication is required to enable them to summon help. In addition, one or more of the following facilities should be provided for such persons:
   (a) means of exit;
   (b) means of de-energizing the apparatus; or
   (c) a shielded area.
5. Audible or visible signals (or both) should be given when the apparatus is about to be energized, and a different signal while the apparatus is energized. Where a γ-ray source is used, 'energized' means that the source is out of its shielded storage location and 'de-energized' means that it has been returned to its storage location.

The main industrial radiographic procedure is the application of X-rays in the non-destructive testing of products, process plant and civil engineering structures. In other applications, particularly where the size of the object requires more penetrating radiation, radiography is undertaken using sealed γ-ray sources, normally cobalt-60 or iridium-192.

Radiographic testing of products is normally part of the production process and is usually undertaken in purpose-designed enclosures with adequate shielding and appropriate safety systems to protect operating personnel. In other situations, such as the testing/inspection of process plant or civil engineering structures, radiography may need to be carried out in conditions that are far from ideal, such as on a construction site. Historically, the main problems have occurred when radiography is undertaken under site conditions, usually by contractors or subcontractors. Doses to radiographers have been quite high, and there have been cases of inadvertent exposure of other workers. There have also been incidents involving lost or broken sources. Recent changes in regulations

have been aimed at ensuring that on-site radiography is undertaken only when it is impracticable to move the item into a proper shielded enclosure and that where on-site work does need to be undertaken it is properly planned and controlled. Before any radiography is performed under site conditions, a risk assessment needs to be undertaken to identify all the risks associated with the proposed work, including non-radiological risks. All reasonable measures need to be taken to protect others on the site, such as applying local shielding and physical restrictions on access. The possibility of accidents that could lead to increased radiation exposure needs to be considered, and measures to prevent such accidents or reduce their consequences should be identified.

13.2.7 Protection in research applications

The two main applications of X-rays in scientific research are **X-ray diffraction (XRD)** and **X-ray fluorescence (XRF) spectrometry**.

Crystals are regular arrangements of atoms and it has been found that definite patterns of scattered radiation, known as **diffraction patterns**, result from irradiating crystals with X-rays. The precise nature of the pattern provides important information on the structure of the crystals. In crystallography, very high-intensity X-ray beams of small cross-section are used; these pose special health physics problems.

In X-ray spectrometry substances are irradiated by X-rays and, as a result of absorbing energy, are excited and emit secondary X-rays. The energy of this secondary radiation is characteristic of the element producing it and so measurement of the secondary X-ray energies enables the substances to be analysed.

The precautions to be observed in this type of work include:

1. The apparatus should be adequately shielded and, where access to the inside of the apparatus is necessary, either the machine must be automatically de-energized or effective steps must be taken to prevent the insertion of any part of the body into the beam.
2. Where a camera or slit-collimating system is in use, the useful beam should be fully enclosed to provide adequate shielding.
3. Automatic visible or audible warning devices should operate whenever the apparatus is energized.

13.2.8 X-rays in other industrial applications

There are many other industrial uses of X-ray radiation sources that need radiation protection control measures to a greater or lesser extent. For example, X-ray inspection units are used on production lines in the food industry in order to check that products, such as bottles of tomato sauce, do not contain any solid contaminants.

X-ray fluorescence analysers are used to examine products and samples in many applications such as sorting for scrap metal recycling, for quality control in foundries, for geochemical surveys in the mining industry, and the assessment of paint and other artefacts in the fields of archaeology and art. Many of these XRF devices are hand-held units (e.g. Niton®) and their portability presents specific radiation protection challenges. Each application of the device must be subject to a prior risk assessment to determine the extent of the required radiation-controlled area and to provide an estimate of the dose rates to which workers and other persons in the vicinity might be exposed. During its use the area within a 2-metre radius of the head is usually designated as 'controlled'. Operation

is undertaken at arm's length and must not be directed towards any persons. Radiation warning signs must be clearly displayed during use and, when the unit is not in use, it must be stored securely so as to restrict unauthorised operation. Operator training is extremely important as is the annual service, maintenance and quality control tests of such equipment. The quality control tests include such things as regular checks for possible tube leakage.

X-ray machines are also used in the **security scanning of baggage** at airports and of parcels in the postal service. Dose rates outside the scanning area are very low, typically 0.5 μGy/hour when the lead-lined curtains covering the entry and exit ports are closed. Just inside the curtains, or when they are parted by a luggage item going in or out, the dose rate can rise to about 60 μGy/hour. However, the units are fitted with a guard that is at least 40 cm long to prevent anyone standing immediately adjacent to the curtain. Items that go through the scanner are typically exposed to a dose of <2 μGy per scan.

A recent technological development, used to detect items such as weapons and narcotics concealed on people at airports and other establishments, is the whole-body X-ray backscatter scanner. The effective dose received by someone having one scan is typically 0.02 μSv or less. This is a very small dose and is comparable to the additional dose received by an airline passenger travelling at cruising altitude for less than 2 min.

13.3 SEALED SOURCES

13.3.1 General principles of protection

Sealed (or closed) radioactive sources are used for a whole host of differing applications in a variety of different industries. These range from the small 'cup' sources, the activities of which are typically only a few tens of kilobequerels, which are used for teaching in schools and colleges, up to sources with activities of many thousands of terabequerels, which are used in irradiators for sterilization purposes. Table 13.3 is an extract from the

Table 13.3 Typical sealed sources used in the non-nuclear industry

Practice	Radionuclide	Activity (TBq)
Irradiators – sterilization and food preservation	Cobalt-60	1.5×10^{5}
	Caesium-137	1.1×10^{5}
Industrial radiography	Cobalt-60	2.2
	Iridium-192	3.7
Calibration facilities	Cobalt-60	0.74
	Americium-241	0.37
Level gauges	Caesium-137	0.19
	Cobalt-60	0.19
Well logging	Californium-252	1.1×10^{-3}
	Caesium-137	7.4×10^{-2}
Moisture/density gauges	Americium-241/Beryllium	1.9×10^{-3}
	Caesium-137	3.7×10^{-4}
Static eliminators	Polonium-210	1.1×10^{-3}
Tritium targets	Hydrogen-3	0.26
Teaching demonstrations in schools	Strontium-90	3.3×10^{-7}

IAEA-TECDOC-1344 ('Categorization of Radioactive Sources') and lists a number of applications and the sources they employ.

Regardless of which sealed source, radionuclide or activity is involved there are certain general radiation protection principles that must be applied. These include:

1. The use of any source must be subject to a prior risk assessment, which must include an estimate of the likely dose rates in routine operation and in all reasonably foreseeable fault conditions. The prior risk assessment allows control measures, such as controlled areas and enclosure interlocks, to be established and incident contingency plans drawn up.
2. Whether it is fixed at a particular site or incorporated into a mobile piece of equipment, a permit to keep and use any source must be obtained from the relevant national authority, which is the Environment Agency in England. This permit contains a list of conditions of use that must be strictly adhered to. If the user intends to stop using a radioactive source he must formally apply to 'surrender' the permit and assure the authority that the source will be disposed of, or transferred to another organization, safely and responsibly. Detailed records of this disposal or transfer must be retained.
3. Sealed sources must be periodically wipe-tested for leakage. Ordinarily, such tests must be performed by the user at least every two years. Records of wipe tests for leakage must be maintained, including the date of the test, the result, who carried out the test and which testing monitor was used. For very high-activity sources it may be unduly hazardous to wipe-test the actual source, and sometimes it is impractical to wipe the source itself because of the construction of the equipment within which it is housed. In such circumstances the permitting authority may accept that it is sufficient to wipe the outside of the source's shielded container or other parts of the equipment where any leaked contamination is likely to accumulate.
4. All sources must be accounted for. This means that they must be assigned a unique identification number and ideally marked with this number. Accurate and up-to-date records of source locations have to be maintained, and sources must be held securely in appropriately shielded receptacles and store rooms when not in use. Sources and/or the equipment containing them must be appropriately labelled with warning signs that include the radiation hazard trefoil symbol.
5. Regular quality-control checks must be carried out on any engineering controls or other safety measures. For example, warning lights that indicate a source is out of its shielded container must have their bulbs periodically tested. Records of these tests must be maintained.
6. Any loss or theft of a source must be notified immediately to the permit-granting authority and, in many cases, also to the police.
7. Workers involved with equipment containing higher-activity sources are usually required to wear some sort of personal dose monitoring device, e.g. a film badge or electronic personal dosimeter (EPD).

Many sources have such high activities that they are subject to extra regulations that reflect the current heightened environment of security against terrorism. In Europe, for

instance, Council Directive 2003/122/Euratom on the control of high-activity sealed sources and orphaned sources was transposed into UK law as the so-called HASS (High Activity Sealed Radioactive Sources and Orphaned Sources) Regulations in 2005, which have since been incorporated into the Environmental Permitting (England and Wales) Regulations 2010 (see section 17.6).

13.3.2 Portable moisture/density gauges

Moisture/density gauges are small, portable industrial gauges (see Fig. 13.4). They contain the sources, detectors and electronics necessary for the measurement. The sources are physically small in size, typically a few centimetres long and a couple of centimetres in diameter. A neutron source (e.g. americium-241/beryllium) is used to assess the moisture content of, say, tarmac. In addition, there is a gamma source (e.g. caesium-137) at the end of a rod that can be inserted into a borehole to determine the density of the material under examination.

The small size of the device makes it susceptible to loss or theft. Also, on busy construction sites, it is not uncommon for these devices to become damaged. There have been a number of cases in which heavy vehicles have inadvertently driven over the gauge and crushed it so that the sources were exposed. This means that principles 3, 4, 5 and 6 in the above list are particularly important when using such portable devices.

13.3.3 Industrial radiography equipment

Although physically small and portable, these devices are usually very heavy owing to the amount of lead or depleted uranium needed to shield the high-activity sources they contain. The sources themselves are similar in size to those in density gauges but typically a thousand times more active. They are usually attached to the end of a specially designed wire that, by electromechanical means, can deploy the source from its shielded container down a tube to whatever needs to be radiographed, for example a welded pipe joint. A

Figure 13.4 A typical nuclear density gauge (NDG) (courtesy of Humboldt Scientific, reproduced with permission).

number of incidents have occurred as a result of the failure of the source to retract properly into its shielded container. The consequences have included overexposure of workers, loss of sources and, sometimes, overexposure of members of the public who have found the sources.

13.3.4 Well-logging devices

This equipment is generally found in areas where exploration for minerals is occurring such as coal mines, oil rigs or gas platforms. The sources, usually gamma emitters such as caesium-137 and neutron emitters such as californium-252, are ordinarily contained within a long (1–2 m), thin (< 10 cm diameter) device that is designed to operate down a borehole. These devices need to be extremely rugged to withstand the harsh environments in which they are used.

13.4 UNSEALED SOURCES

Unsealed, open or dispersible sources can be either liquid, gaseous or in the form of a powder. They are commonly used in industry and research for the purpose of conducting so-called 'tracer' studies, i.e. the open sources are released into some environment and then used to follow some physical, chemical and/or environmental process. The radioisotopes used are chosen to be easily detectable in samples even when the concentration is very small, i.e. when 'trace' amounts are present. For example, tritiated water (T_2O) can be used to follow groundwater movements in geological surveys, and a range of radioisotopes, such as phosphorus-32, carbon-14 and sulphur-35, can be used to label molecules which can then be used in biological studies.

Unsealed sources are also used in medical diagnostics and therapy and these applications are dealt with in section 14.5.

As with all uses of unsealed radioisotopes, the problems of protection arise from the spread of radioactive contamination and the risks of internal doses to workers and members of the general public (see Chapter 9). Care should be taken to ensure that the radioisotopes are appropriately accounted for, although use can be made of records produced for waste decay and disposal purposes to support the accountancy arrangements.

It is worth noting that many industrial processes expose workers to NORMs that are unsealed. These include dusts in uranium mining, and the sludges and scales of the oil industry.

SUMMARY OF KEY POINTS

X-rays: electromagnetic radiation; originate from changes in atomic electron energy levels or as bremsstrahlung when electrons strike high atomic number target material.

X-ray equipment: tube and a separate control unit.

Tube: cathode and anode in evacuated glass tube.

Control unit: high-voltage supply controls the quality of the X-rays; the low-voltage supply to the filament defines the intensity.

Protection against X-rays: X-ray sets can be switched off but are hazardous when operating. Safety measures depend on the particular application but generally involve a

combination of staff training, shielded rooms, filtration, warning systems, personal and area monitoring.

Site radiography: should only be carried out when it is impracticable to move an item into a properly shielded enclosure. Full risk assessment, local shielding and physical restrictions on access are normally required.

Monitoring of radiographic installations: radiation surveys must be comprehensive and recognize the possibility of scattering into adjacent areas. There are unique problems associated with personal monitoring because of the small cross-sections of X-ray beams.

XRD and XRF: X-ray diffraction and X-ray fluorescence are research or analysis techniques that use X-rays. X-ray fluorescence analysers are often hand-held devices and are used in industrial applications such as scrap metal recycling, quality control in foundries and geochemical surveys in the mining industry.

Sealed sources: sources containing any radioactive substance whose structure is such as to prevent, under normal conditions of use, any dispersion of radioactive substances into the environment. They are used in a wide variety of industrial applications with a huge range of activities.

Leak testing of sealed sources: sealed sources should be tested at regular intervals (in the UK this is generally every 2 years) using an appropriate method to detect leakage of radioactivity from the source (see section 10.3 for more details).

Unsealed sources: dispersible sources, which can be gaseous, liquid or powder. They are commonly used as 'tracers' in many industrial applications, as well as in medicine (see section 14.5).

REVISION QUESTIONS

1. Describe, with the aid of a sketch, the operation of an X-ray tube.
2. Compare the effect of varying the tube voltage and the tube current on the radiation output.
3. Compare the radiological hazards posed by large γ-sources and X-ray sets.
4. Design an enclosure for the routine X-radiographic examination of large metal castings. Indicate the safety features that are included.
5. What special problems arise in the monitoring of X-rays?
6. Prepare a justification case for the use of X-rays for whole-body security scanners at airports.
7. Write a risk assessment for the installation and use of a 60-TBq Cs-137 source for blood irradiation at a research facility.

14 Radiation protection in medicine

14.1 APPLICATIONS

Ionizing radiation is a powerful tool in many branches of medicine, as both an aid to diagnosis and a means of therapy (treatment). For diagnostic purposes there are two basic approaches. The first is to pass a beam of radiation, normally X-rays, through the body onto an image-capture device such as a semiconductor flat panel detector (FPD) built into the patient table. The different degrees of absorption in the body produce a picture that gives information on the **structure** of the internal organs. For example, conventional X-ray techniques can reveal broken bones, diseased lungs or the presence of a tumour. Another technique of great importance in medical diagnosis, based on the same principle, is **computed tomography** (**CT**), popularly known as **scanning**. A tomograph is an image of a section, i.e. a slice, through an object, in this case the human body. The second diagnostic approach is to introduce a **radioactive tracer** into the body, for example by mouth or by injection into the bloodstream, and to observe its behaviour by means of external detectors. This technique can give information on the location and development of disease and **functioning** of body systems, such as cerebral blood flow.

The main therapeutic application of radiation is in the treatment of cancer. Radiation can induce cancer and yet, paradoxically, in some cases it can also cure the disease. This is because cells that are dividing rapidly are particularly sensitive to radiation and, as cancers are groups of cells dividing in an uncontrolled manner, it follows that they are often more sensitive to radiation than normal cells. As with diagnosis, radiation therapy procedures can involve the use of beams of radiation to target the diseased tissue, or the use of radioactive materials injected into or applied to the body. Another technique, known as **brachytherapy**, involves the application of small, sealed sources directly onto the site of the cancer. Clearly, the doses or levels of radioactive material involved in therapeutic procedures have to be very much higher than those used in diagnostic applications.

Those applications that involve the introduction of radioactive substances in liquid form to the body, whether for diagnosis or treatment, are generally referred to as **nuclear medicine** techniques. The dispensing, handling and application of the radioactive preparations, especially with the high levels of dose needed in therapeutic procedures, can give rise to radioactive contamination, and so appropriate control procedures need to be applied. In particular, special measures are needed when dealing with patients to whom radioactive materials have been administered.

Techniques involving radiation and radioisotopes are of great value in medical diagnosis and treatment, but it must always be borne in mind that the resulting radiation exposure involves risks that need to be weighed against the potential benefits.

All medical procedures involving radiation have to be shown to be justified and then optimized by appropriately trained medical staff so that the benefits to individual patients outweigh any detriments. It is worth noting that, as well as exposure of the patients undergoing their own diagnosis or treatment, the use of radiation sources in medicine can give rise to the exposure of medical staff and the public, and the contamination of the environment. In addition, volunteers involved in biomedical research may be exposed to ionizing radiation, as can comforters and carers of patients, and asymptomatic persons taking part in a health screening programme, e.g. in breast cancer screening. International Commission on Radiological Protection (ICRP) Publication 105 ('Radiological Protection in Medicine') provides further detail on the exposure of these unique medical cohorts.

Medical exposure is by far the largest man-made contributor to population dose (98 per cent) and, in addition, there are large numbers of medical staff who could potentially be exposed to ionizing radiation. Consequently, dose optimization and patient dose-reduction strategies can have a significant effect on the collective dose to society.

14.2 GENERAL PRINCIPLES AND ORGANIZATION

Special problems arise in radiation protection in medicine because the well-being and reassurance of the patient is of prime importance. Also, a patient who has been given a large intake of a radioisotope may represent a significant radiation hazard to others, not only during their hospital stay but also at or on their way home. Very often the normal methods of protection described in earlier chapters such as shielding, distance and containment cannot be applied to patients in the usual way. However, with common sense, the patient can be adequately cared for without excessive risk to others.

The organization and responsibility for radiation protection in medical establishments vary from country to country. In the countries of the European Union (EU), national regulations are based on the general principles set out in Council Directives. In relation to protection of the patient, the relevant Council Directive is 97/43/Euratom of 30 June 1997 (the 'Medical Exposure' Directive). The important principles of this Directive are summarized below.

1. Medical exposures should be justified by showing that they may be expected to produce a net benefit. This process of justification applies at two levels:
   (a) generic demonstration (normally at national level) that any new type of medical procedure is justified before it is introduced, and
   (b) the application of the procedure to an individual patient should be shown to be justified, taking into account the objectives of the exposure and the particular circumstances of the patient.
2. All medical exposures should be shown to be optimized. In particular:
   (a) in procedures undertaken for diagnostic purposes, the level of exposure should be as low as practicable, consistent with obtaining the required information, and
   (b) in the case of exposures for therapeutic purposes, the exposure should be individually planned and should ensure that the doses to regions outside the target volume are as low as reasonably achievable.

3. Optimization is also taken to mean that, in addition to careful planning, the best available techniques should be applied and that the whole process should be subject to quality assurance.
4. Procedures involving radiological exposure of patients should be undertaken in accordance with written procedures and protocols.
5. Clinical audits (including patient dose audits) should be undertaken periodically to confirm the effectiveness of the procedures and protocols. These can be undertaken at the level of individual departments, medical establishments or at national level.
6. Responsibilities should be clearly defined. For example, in many cases, a medical professional (the referrer or prescriber) will refer a patient for a procedure to another medical specialist (the practitioner), who will decide on the details of the radiological procedure. A radiographer or a medical physics technologist (the operator) will then carry out the actual procedure, with support from medical physicists, particularly with respect to optimization.
7. All those involved in the process should have adequate theoretical and practical training and should hold appropriate formal qualifications, diplomas or certificates.
8. Special attention needs to be given to situations where there is a request for a radiological procedure involving a child, a pregnant woman or a breast-feeding mother, or where the techniques deliver high radiation doses to patients, for example computed tomography. Special attention also has to be given to exposures where there is no direct benefit to the exposed individuals, such as volunteers taking part in research studies.
9. All radiological procedures should be performed in such a way as to minimize the dose to other persons.

The requirements for protection of workers in the medical field and of others who might be exposed as a result of medical procedures are essentially those that apply to any other industry and are set out in Council Directive 96/29 Euratom.

Each EU member state complies with the relevant Directives by means of its own internal legislation and regulations. In the UK, for example, all work with radiation is subject to the requirements of the Ionizing Radiation Regulations (1999), under the Health and Safety at Work Act (see section 17.5). These apply to workers and to members of the public exposed as a result of the employer's activities. An additional set of regulations, the Ionizing Radiation (Medical Exposure) Regulations (2000), applies to patients. Under both sets of regulations the ultimate responsibility for radiation protection lies with the chief executive of the hospital, who is deemed to be the 'employer'.

As in any other industry, the radiation employer must formally appoint a competent and suitable radiation protection adviser (RPA). In some situations the RPA may be an external consultant covering several hospitals and, as such, he or she may not always be on site. Nevertheless, it is a legal necessity for the employer to appoint one or more radiation protection supervisors (RPSs) who will closely supervise radiation safety on a day-to-day basis in every relevant department within the hospital.

In addition to the requirement of the Ionizing Radiation Regulations (1999) that the radiation employer appoint an RPA, the Ionizing Radiation (Medical Exposure) Regulations

(2000) require a medical physics expert (MPE) to be appointed for patient protection. The duties of the MPE, who may be the same person as the RPA, include providing advice on optimizing radiation doses to patients.

In the case of patient protection, as set out above, responsibilities are separately defined for the referrer, the practitioner and the operator. The referrer (sometimes known as the 'prescriber') is a healthcare professional, such as a doctor in general practice, who is entitled to refer individuals for medical exposures to a practitioner. The practitioner, as defined in the EU Medical Exposure Directive (97/43/Euratom), is the person who takes clinical responsibility for the exposure and justifies the use of ionizing radiation on a case-by-case basis. Typically, he or she is a radiologist or an oncologist. The operator is a person who effects any practical aspect of the exposure, for example the radiographer who actually operates the X-ray machine or the radiopharmacist who prepares the isotope for injecting into a patient.

The key issues with regard to safety in medical situations are the initial and ongoing training of staff and the conduct of all radiological procedures within a strict quality assurance regime. This must ensure, among other things, that there is a clear chain of responsibility, that patients undergoing procedures are correctly identified, and that the procedure is appropriately optimized and conducted to give maximum benefit to the patient. It must also ensure that staff and others supporting or caring for the patient (such as family and friends) are adequately protected. It is emphasized that special consideration should be given before radiological procedures are applied to children or pregnant women. In the case of women who could be pregnant, the timing of procedures should take account of the last menstrual period. The application of nuclear medicine procedures, i.e. involving the administration of radionuclides, needs special attention if the patient is breast-feeding.

14.3 DIAGNOSTIC PROCEDURES

14.3.1 Diagnostic radiography

Diagnostic X-ray imaging is very common in modern medicine. For example, in the UK, over 40 million radiological examinations are performed each year. The great majority of these are conventional medical and dental X-rays but increasing numbers of more sophisticated tests, such as CT scans, are being undertaken. They may be carried out as part of the investigation of symptoms in an individual patient or as part of a general screening process. In the latter case, the tests are known as **asymptomatic** because they are not undertaken in response to reported symptoms but as part of a programme for early detection of certain medical conditions. An important example of this is mammography as part of a breast screening programme, which is made available to women in the 50–70 year old age group in order to detect early signs of breast cancer.

Most diagnostic radiographic examinations are performed using X-ray sets in combination with some type of digital image acquisition system. As a result of recent technological developments, almost all 'film–screen' systems have been replaced by **digital radiography (DR)** or **computed radiography (CR) systems**. These use solid-state detection systems and have the advantage that the resulting image is stored in electronic format and can be accessed remotely by medical staff. Digital data files allow for image manipulation, processing and analysis, which were not possible with film-based detectors. This can lead to improved diagnosis, and provides opportunities for patient dose reduction.

Whichever technique is used, careful selection of the X-ray beam quality (voltage) and the exposure enables good-quality radiographs to be obtained with quite small doses to the patient. For example, using the latest technology and the best techniques available, a chest X-ray can deliver as little as 150 μGy entrance surface dose to the chest of the patient (which gives an effective dose of only 10 μSv). With properly designed and operated systems, and with the minimum practicable beam size, the dose to other parts of the body will be much less than this. In some cases it may be necessary to take more than one radiograph, but clearly the number of 'shots' should be kept to the absolute minimum.

The dose to the radiographer is minimized by good design of the facility, for example by the provision of a shielded cubicle in which the radiographer must stand to operate the set. Occasionally, difficulties arise; for example, young children may need to be held in the correct position. If a harness cannot be used it is better for the parent rather than the radiographer to hold the child, as the parent is unlikely to be exposed frequently in this way. A similar problem sometimes occurs in dental radiography when it is not possible to clamp the film in position in the mouth. In this case, the patient should hold the film rather than the dentist or other members of the staff.

In addition to the hazard from the primary beam, X-rays are scattered from the patient or nearby materials, so constituting a further hazard. This scatter needs to be considered when deciding how to protect the staff or members of the public who might be involved.

An important point to bear in mind in medical X-ray work is that a significant reduction in dose can be obtained with quite thin shielding because of the relatively low X-ray energy used (often less than 100 kVp). For example, lead-impregnated materials are available, which can be made into aprons and gloves and are equivalent in shielding ability to approximately 0.25–0.35 mm of lead, which is suitable to protect staff from scattered radiation. Similarly, the lead-glass of control room windows is equivalent to 2 mm of lead and is sufficient to protect radiographers from the primary X-ray beam.

14.3.2 Diagnostic fluoroscopy

In **fluoroscopy**, the detection system is a fluorescent screen coupled to an image intensifier. Rather than the single short-duration pulse used in radiography, the X-ray tube remains on (or is continuously pulsed) during the examination. The screen fluoresces under irradiation and therefore gives a live picture. The principle of this technique is illustrated in Figure 14.1.

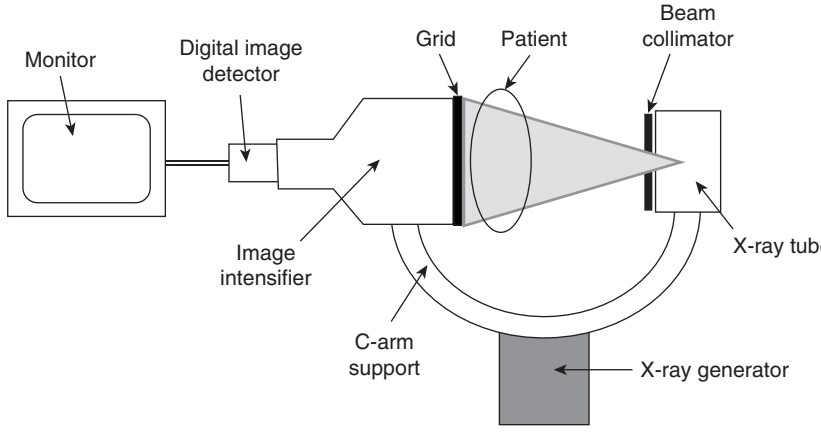


Figure 14.1 Principle of fluoroscopic examination.

The output from the image intensifier can be fed to a video system allowing the medical staff to view the moving images on a television monitor, which is often outside the controlled radiation area. In 'digital' fluoroscopic systems the analogue video system can be digitized with an analogue-to-digital converter. Alternatively, digitization may be accomplished with a digital video camera (charge-coupled device) or via direct capture of X-rays with a flat panel detector (FPD) similar to the type used in modern radiographic systems.

In some types of examination, much higher-quality images can be obtained, often with reduced dose, by injection of contrast media. These are chemical solutions that absorb X-rays more effectively than the body organs or fluids and so give enhanced images. This technique is commonly used in angiography, which is concerned with investigations of blood vessels.

Fluoroscopy is also used during interventional procedures so that, for example, a surgeon can view procedures being undertaken inside the body of the patient. A modern fluoroscopic facility is shown in Figure 14.2. From a staff protection viewpoint, an important consideration is that the surgeon's hands may be close to the X-ray beam for appreciable periods of time and the resulting 'extremity' doses need to be monitored and controlled. In addition, a significant amount of radiation is back-scattered from the patient, and medical staff within approximately 2 metres of the patient must wear lead aprons and, if required, lead thyroid shields. It might also be necessary for them to make use of lead-glass spectacles or a lead-glass screen to protect their eyes.

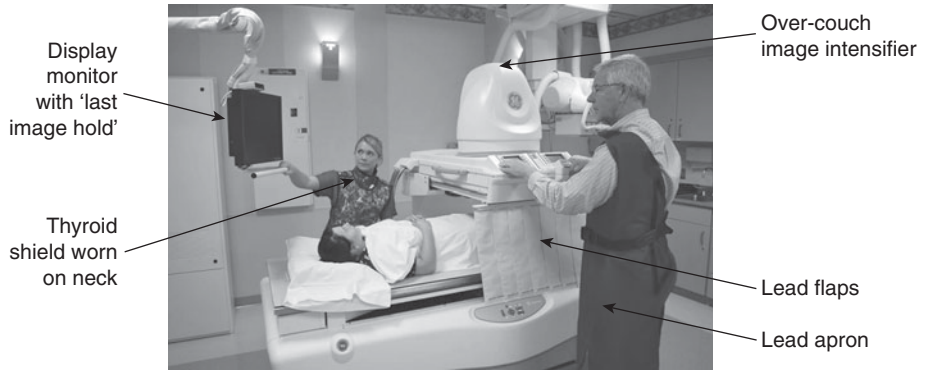


Figure 14.2 The use of fluoroscopy equipment. Reproduced from Huntingdon Daily.com

The annual occupational effective dose to interventional radiologists and cardiologists is typically 2 mSv, although this can be significantly higher depending on their workload and the radiation protection control measures they employ. In some cases, the radiation exposure of this group may exceed three-tenths of the annual dose limit and make it necessary for them to be designated in the UK as classified persons for occupational dose monitoring purposes.

14.3.3 Computed tomography

Computed tomography (CT) uses an X-ray tube and an array of detectors arranged in a supporting framework to rotate around the patient. A continuously rotating collimated X-ray beam passes through the body, and the output from the detectors is analysed by a computer, which produces pictures of cross-sections, or slices, of the body. As in the case

of fluoroscopy, the quality of the images can be greatly increased by injection of contrast media. The principle of operation is illustrated in Figure 14.3, which shows a CT system in which the source and detector system are rotated around the patient as he or she is traversed through the system. A typical modern installation is shown in Figure 14.4. CT is used for many types of radiological examination and is particularly useful for the diagnosis and follow-up of malignant tumours. Modern CT equipment is capable of multi-slice helical scanning where the patient table, and therefore the patient, moves through the X-ray fan beam while the tube is rotating. Rather than just a slice, or set of slices, a 'volume' of the patient is irradiated. This leads to a much faster acquisition of imaging data from a greater section of the patient but, if not controlled properly, may lead to increases in patient effective dose.

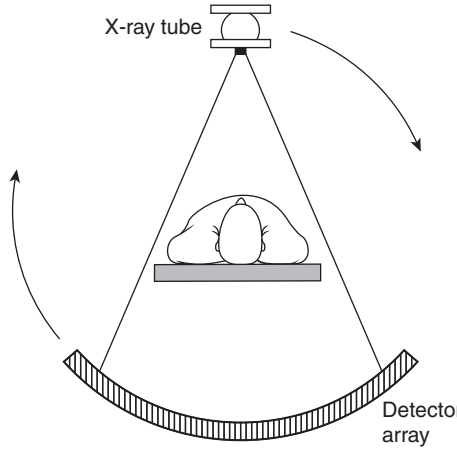


Figure 14.3 Schematic illustration of transmission computed tomography.

Figure 14.4 A modern computed tomography installation (courtesy of Philips Healthcare).

Computed tomography scanning achieves high-contrast resolution by using a high X-ray tube output, which in turn leads to relatively high patient doses. The actual dose received by the patient depends on the type and extent of the examination, but typical effective doses are between 1 mSv and 10 mSv. Although CT is a low-frequency technique representing only 7 per cent of all diagnostic X-ray examinations carried out annually in the UK, its contributes nearly 50 per cent of the total UK collective dose from diagnostic X-ray procedures.

The equipment is normally located in a shielded room with the radiographer located in an adjacent control room with a lead-glass viewing window. In the event that the patient needs attention, the X-ray beam would automatically switch off when the door is opened.

Another type of tomographic technique used in diagnosis is **positron emission tomography** (**PET**), which involves injection of radioisotopes into the body. This is discussed later in section 14.5.

14.4 RADIOTHERAPY

It has been noted that the main application of radiotherapy is in the treatment of cancer. The aim is to deliver as high a dose as possible to the malignant tissue without causing excessive injury to surrounding healthy tissue. Typically, absorbed doses of a few tens of grays are required and they are usually delivered as a series of smaller doses, for example 20 doses of 2 Gy at intervals of 2 or 3 days. This fractionation is necessary to reduce deterministic side-effects.

In **teletherapy**, or **external beam therapy**, the radiation is administered by a machine positioned some distance from the patient. The most common method of treatment uses equipment such as linear accelerators to deliver high-energy electron beams of 6–20 MeV or high-energy X-rays of 6 MVp. However, in some countries, collimated beams of γ radiation from large Co-60 sources are still used. For treatment of superficial tissue, X-rays of about 200 kVp are often used.

In addition to selection of the appropriate energy, the dose to healthy tissue is minimized by varying the direction of the beam through the body. This is done either by using a different orientation for each treatment or by continuously rotating the source around the tumour during the treatment. The principle is illustrated in Figure 14.5, which shows treatment of a brain tumour using a rotating teletherapy unit containing a linear accelerator. Although the tumour is being irradiated continuously, the surrounding regions are exposed for only a small fraction of the time. It is essential to use a well-defined beam, and this is achieved by means of **collimators**. Modern systems have collimators that incorporate many overlapping and independently movable 'leaves', called multi-leaf collimators (MLCs). Intensity-modulated radiation therapy (IMRT) is an advanced external beam technique used to minimize the amount of normal tissue being irradiated. The radiation beam intensity modulation is achieved by moving the leaves in the MLC during the course of treatment, thereby delivering a radiation field with a non-uniform (i.e. modulated) intensity.

Except in some low-voltage (< 100 kVp) superficial X-ray therapy, the problem of providing local shielding is such that the treatment must be performed in a shielded room with interlocks arranged to shut down the equipment should the door be opened. In so-called **'mega-voltage' therapy**, such as that carried out with linear accelerators (or 'linacs'), the equipment is housed in a room with no windows and concrete walls that are over a metre thick. Entry into the room is often via a shielded maze, which is designed to reduce scattered

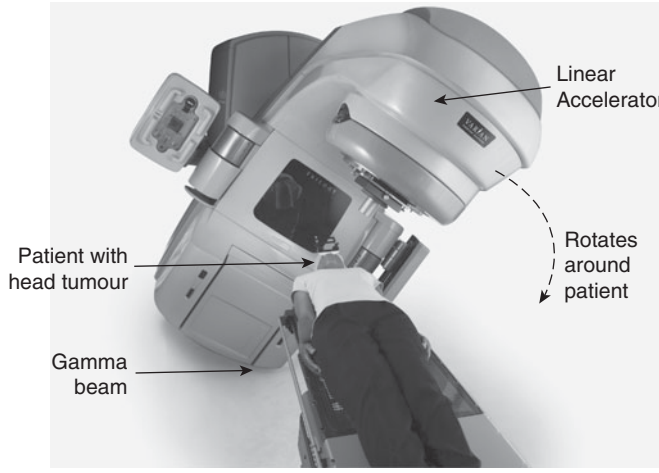


Figure 14.5 Treatment of a brain tumour using a linear accelerator teletherapy unit. (Image courtesy of Varian Medical Systems of Palo Alto, California. Copyright (2012), Varian Medical Systems. All rights reserved.)

radiation to outside areas. The radiotherapist remains outside the room, observing the patient via closed-circuit television (CCTV) and communicating via an intercom system.

All radiotherapy is carried out within an overall system of quality assurance. This requires a detailed treatment plan for the patient and the application of suitable quality controls at all stages. Usually, this involves a simulation system that includes a CT scan in order to define precisely the region to be irradiated. Often, a mould is produced for a patient and this serves as both a patient immobilizer and a means of defining beam direction. The timing of exposures is under automatic control to ensure the correct dose to the target area. Regular testing and calibration of equipment (often daily) is a key aspect of quality assurance.

In addition to gammas or electrons, beams of protons can be used to treat cancer. Protons with energies ranging typically from 70 to 250 MeV can be produced from particle accelerators such as cyclotrons or synchrotrons. The advantage of proton therapy is the ability to localize the radiation dosage more precisely when compared with other types of external beam radiotherapy.

As noted earlier, radiotherapy can also be effected by **brachytherapy**, which involves the application of small, sealed sources into the tumour (interstitial brachytherapy) or adjacent to a tumour (intracavitory or contact brachytherapy). The sources are normally either applied by surface applicators or inserted into body cavities or organs by specially designed delivery systems called remote afterloaders. In these cases, the exposure is fractionated, with individual exposures lasting from a few minutes to a few hours. In some cases, for example for treatment of prostate cancer, small radioactive pellets or 'seeds' are surgically implanted and remain in the body delivering a dose at a relatively low rate until the required dose has been delivered. The most common types of source used in brachytherapy are iridium-192, caesium-137 and cobalt-60.

The source delivery systems are designed to minimize radiation exposure of staff. Where seeds are implanted into patients, special attention has to be given to the control of exposure of nursing and medical staff and controls need to be placed on visitors. Adequate protection can be achieved by sensible application of the principles of time, distance and shielding. In the case of sources that are reused, regular leakage testing is required

and written emergency procedures should specify the actions to be taken in the event of damage to or loss of a source.

It is possible, in some cases, for patients containing sources to be discharged. This is decided on a case-by-case basis, taking account of the radionuclide, the half-life and the dose rate, which together define the risk to other persons.

Also, in radioisotope therapy, cancers can be treated using unsealed radioactive materials. For example, many gigabequerels of iodine-131 can be administered orally to patients in the radiotherapy of thyroid cancer. The use of unsealed radioisotopes in medical diagnosis and therapy is dealt with in section 14.5.

Finally, it must be re-emphasized that when external beams of radiation are used, or sealed sources or radioactivity injected into the body, the aim in radiotherapy is always to deliver a precisely predetermined dose to the target region while minimizing as far as possible the dose to adjacent healthy tissue.

14.5 NUCLEAR MEDICINE

The term **nuclear medicine** refers to the introduction of radioisotopes in liquid (or occasionally gaseous) form into the body for either diagnostic or therapeutic purposes, or for the study of disease. The unsealed radioactive material is administered orally, intravenously or by inhalation of gases. The scale of application of these techniques is much less than for external radiation beam procedures; nevertheless, they are still commonly used practices in health care. For example, in the UK in the year 2004, some 700 000 nuclear medicine procedures were undertaken, of which about 98 per cent were for diagnostic and 2 per cent for therapeutic purposes.

In nuclear medicine, special attention should be given to women who are breast-feeding. Depending on the procedure involved, it may be necessary to advise the patient to cease breast-feeding until it is established that the risk to the child is sufficiently low. Precautions might also need to be taken to protect relatives, friends and others who come into contact with patients, particularly when they are discharged from hospital while still retaining radioactive material.

14.5.1 Diagnostic radioisotope tests and nuclear medicine imaging

The purpose of radioisotope diagnostic tests is the investigation of body function. By introducing radioactive tracers in a suitable chemical form into the body and observing their behaviour using external detectors, or by monitoring excretion, important information on the functioning of body organs may be obtained. The pattern of distribution of the radioactive tracer can be constructed into an image by a **gamma camera**, which consists of a collimated scintillation detector coupled to an array of photomultiplier tubes (see Fig. 14.6).

This basic scintigraphy technology has now been developed to include **single photon emission computed tomography (SPECT)**. As the name implies, this is very similar to transmission CT except that the system detects γ-ray photons emitted by the radioactive tracers in the body and constructs an image of a section through an organ or the whole body using one, two or even three gamma camera detector arrays or 'heads'. The organs that can be imaged by this technique include the lungs, brain, liver, spleen, kidneys, thyroid, bone and blood. Most of these tests use suitable pharmaceuticals labelled with a radionuclide (called **radiopharmaceuticals**), commonly technetium-99m (Tc-99m). The great advantage of this is that it can be obtained from a radionuclide generator. The

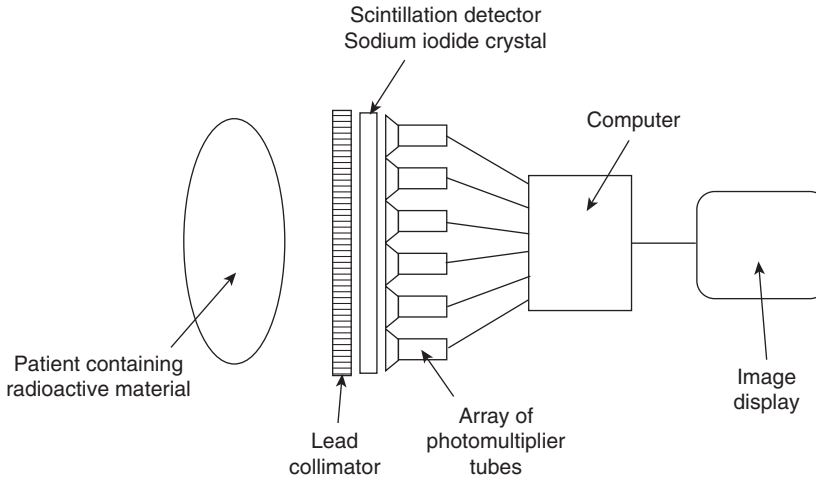


Figure 14.6 Schematic illustration of gamma camera used in nuclear medicine imaging.

generator typically contains 0.04 TBq of molybdenum-99 (Mo-99), which has a half-life of 66 h and decays to the pure γ-emitter Tc-99m, which has a half-life of 6 h. The Mo-99 is absorbed onto tin dioxide and, as the Tc-99m daughter is produced, it is released into saline solution in the generator. The saline solution containing the Tc-99m is eluted into phials and, if necessary, combined with pharmaceuticals in preparation for administration.

Another technique used is **positron emission tomography** (**PET**). In this case, the radionuclide tracer is a positron emitter, usually fluorine-18, and the detection system detects the 0.51 MeV annihilation γ-rays. These require greater shielding than the softer γ radiation from some other radionuclides. In addition, owing to the very short half-lives of the radioisotopes involved, such as 110 minutes for fluorine-18, facilities that offer PET imaging generally require an on-site accelerator called a 'cyclotron' to produce the required radiopharmaceuticals. The installation and operation of accelerators such as cyclotrons represent additional radiation protection challenges.

Less sophisticated, non-imaging, techniques are also used. For example, a single scintillation detector placed close to the thyroid can be used to study the functioning of this organ (see Fig. 14.7).

The quantities of radionuclides involved in these tests range from tens to hundreds of MBq and the dose to the patient is generally a few mSv. With increasing use of PET scanning, particularly involving the use of F-18, the dose received by staff involved in nuclear medicine procedures requires careful monitoring and control. With appropriate

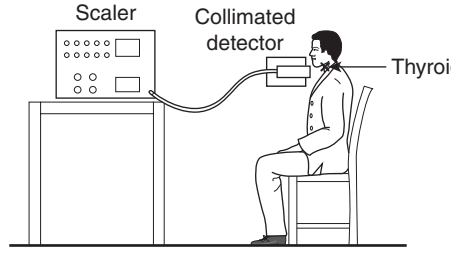


Figure 14.7 Thyroid radioiodine uptake test.

methods of working, fingertip and eye dose can usually be controlled and it is often the whole body dose that is limiting.

Under many circumstances the patient can be discharged immediately after the examination has been completed because the low activities and short half-lives of the radioisotopes involved do not leave a residual activity that would represent a significant hazard to other people. The radioactivity is normally reduced to a very low level within a few days by radioactive decay and excretion.

14.5.2 Radioisotope therapy

In some circumstances radiation therapy is best performed by the ingestion or injection of **radionuclide solutions** into the body. Specific nuclides or radiolabelled pharmaceuticals are chosen, which concentrate in the organs requiring treatment, thus minimizing the dose to the rest of the body. The majority of therapeutic procedures involve the administration of nuclides of fairly short half-life (8 days or less) and the quantity is selected so that the required dose is delivered from the time of administration until the nuclide decays or is excreted. The main applications for radioisotope therapy are the treatment of thyroid cancer and thyrotoxicosis, using iodine-131. Typically, quantities of up to about 5000 MBq are administered for the treatment of thyroid cancer, giving a thyroid absorbed organ dose of up to 100 Gy, and a dose to the whole body (effective dose) of up to 1 Sv. Treatment for thyrotoxicosis, although more common than for thyroid cancer, involves only about one-tenth of the quantity of iodine-131 and therefore one-tenth of the dose.

Patients containing therapeutic quantities of radioactivity should be nursed under conditions that permit easy containment of radioactivity in case of contamination. Ideally, special rooms should be provided with en-suite facilities and all surfaces designed to permit easy cleaning. Where γ-emitters (such as iodine-131) are involved, the room may need to be shielded to ensure that the dose rate in adjacent areas is not significantly increased. The ventilation system should provide an adequate rate of air change, typically 5–10 air changes per hour, and should be designed to ensure that there is no possibility of the air being recycled to other areas. Protective gowns and gloves should be worn when handling the patient, contaminated linen or excreta, and a special storage area should be provided for contaminated linen waste and samples of excreta. The Radiation Protection Supervisor (RPS) should specify any limitations on the time allowable for nursing procedures or visiting periods. Washing and monitoring facilities should be provided for use when leaving the area and regular radiation and contamination surveys should be made of the ward. Patients must only use a so-called 'hot' toilet that is designated for aqueous radioactive waste disposal. Within the hospital, liquid wastes and excreta are normally routed via a dedicated drainage system, but ultimately the discharge is to the normal public sewage system where they are diluted by the much greater quantities of uncontaminated liquid from other areas.

In some cases, such as treatment for thyrotoxicosis, patients undergoing therapeutic nuclear medicine procedures may be treated as outpatients and discharged on the day of treatment. Where the administered quantity of radionuclide is higher, such as for the treatment of thyroid cancer, the patient would normally remain in hospital for a few days during which the level of retained radioactivity would rapidly reduce.

Within a hospital, the source preparation is undertaken in a radiochemical laboratory called the radiopharmacy and protection is achieved by the methods described in Chapters 8 and 9, that is by minimizing the quantities of radioactive material handled, by containing it whenever possible and by the use of good procedures and facilities. The grade of the

laboratory should be appropriate to the radiotoxicity and the quantity of the nuclides in use. It will be recalled that in such laboratories special attention is paid to surface finishes and to ventilation. Fumehoods are an essential feature even if quite low levels are being used. It is very important to have separate facilities for diagnostic and therapeutic work as low-level diagnostic tests can be ruined by cross-contamination from highly active equipment used in therapeutic work.

14.6 CONTROL AND DISPOSAL OF RADIOACTIVE MATERIALS

A large hospital may hold a large inventory of radioactive sources, both sealed and unsealed. One or more special storage areas may be required, located in positions that minimize the risk of fire or flood damage. The importance of keeping good records of the location of each source and the regular mustering of all sources cannot be overemphasized. Leakage tests are required to be performed biennially on all sealed sources and any source showing significant leakage must be withdrawn from use immediately. Sources not in the main storage area should be kept and transported only in approved containers. These containers should be constructed to provide adequate shielding and adequate containment to prevent dispersal in the event of damage to the source. They should be clearly marked **radioactive** and carry the standard trefoil symbol. Written procedures should be available detailing the actions to be taken in the event of loss, breakage or spillage of a source.

Large sealed sources are subject to special requirements to ensure safe and secure retention and to avoid the possibility of accidents or misuse. Disposal of large sources is very costly and it must be ensured that sufficient funding will be available for this purpose at the end of their useful life.

In the UK, hospitals are required, under Schedule 23 of the Environmental Permitting Regulations 2010 (formerly the Radioactive Substances Act 1993), to register their use of radioactive materials and to obtain a permit to keep and use radioactive materials, and to accumulate and dispose of radioactive waste. The general policy on radioactive waste disposal from hospitals is to use conventional local methods whenever possible. Solid wastes containing biological materials (known as clinical wastes) are normally collected in yellow plastic bags or bins and sent for high-temperature incineration. The incinerator ash is disposed of, with other low-level solid waste, at a suitable refuse tip. As discussed earlier, low-level liquid wastes such as laboratory washings or excreta from patients are discharged to the normal sewage system.

Records must be kept of all waste discharges, both solid and liquid. It is not usually practicable to measure the activity and composition of the waste, but sensible estimates can be made from a knowledge of the quantities of the various radionuclides in use.

The disposal of radioactive waste is described in greater detail in Chapter 12.

SUMMARY OF KEY POINTS

Radiation is a powerful tool in medicine and is used in both diagnosis and treatment.

Justification: medical procedures involving radiation should be used only when there is a net benefit to the patient, i.e. the benefits outweigh the detriments.

Optimization: procedures should be optimized to produce the maximum benefit from doses that are as low as reasonable practicable.

Diagnostic radiography involves the use of radiation beams, usually X-rays, to give information on the structure of internal organs. Nowadays, in most applications, digital detectors are employed, which means that the images can be stored on electronic media and accessed remotely.

Fluoroscopy is a means of obtaining moving images of physiological processes and so obtaining information on the functioning of body organs. Image intensification is an essential part of the process if patient and staff doses are to be kept at an acceptable level.

Computed tomography (CT): popularly known as CT scanning. Uses an X-ray fan beam and an array of detectors to build up a series of images of sections of the body. The equipment is located in a shielded room and the radiographer should not normally need to enter the room during a scan.

Contrast media: chemical solutions that normally absorb X-rays more effectively than the body organs or fluids and so give enhanced X-ray images.

Radiotherapy: treatment by means of X-rays, γ-rays, electrons or protons, or small, sealed sources. External beam therapy is carried out in a treatment room specially designed for the purpose. Close attention is needed in the design of shielding, interlocks and warning devices to minimize the doses to staff.

Brachytherapy: involves the application of small sealed sources directly onto the site of the cancer. Care is needed to prevent a patient containing a small sealed source becoming a radiation hazard to other patients and visitors. Areas where such patients are treated may need to be designated as radiation-controlled areas.

Nuclear medicine: the use of radioisotopes in diagnosis and therapy. This requires special attention to the design of radioisotope laboratories and nuclear medicine departments for activities such as the preparation of radiopharmaceuticals and local storage, waste storage and disposal, radionuclide administration to patients, clinical measurements, sample measurements and decontamination. Diagnostic procedures usually use short half-life and low-activity nuclides, but radioisotope therapy involves high levels of radioactivity where extra precautions are necessary in the nursing of the patients in order to avoid staff exposure and the spread of contamination.

Gamma camera: consists of a collimated scintillation detector coupled to an array of photomultiplier tubes. It acquires images of body functions (as opposed to anatomical images) by detecting distributions of radioisotopes previously administered to the patient.

Disposal of waste involves consideration of the permitted route(s) of disposal (e.g. sewers, local refuse collection, etc.), the quantities and activities involved and the records to be kept.

REVISION QUESTIONS

1. What are the main applications of radioactive substances and other sources of ionizing radiation in medicine?
2. List some basic principles that are applied in the protection of staff in an X-ray department.
3. Discuss how radiation protection of the patient can be achieved in nuclear medicine.
4. What protection measures are required when sealed sources in brachytherapy are used?
5. Describe the special radiological problems that may arise in the nursing of patients containing therapeutic quantities of radiopharmaceuticals.

15 Risk assessment

15.1 INTRODUCTION

The objective of radiation protection is to protect people and the environment against the harmful effects of ionizing radiation. In order to achieve this objective it is necessary to estimate the chance that something could go wrong, evaluate the consequences that would result and install sufficient control and protection systems to ensure that these consequences are acceptably low. This is achieved through a process called **risk assessment**, which involves:

- identifying the **hazard**;
- estimating the size of the **risk**; and
- **assessing** its importance in comparison with other risks.

The results of the risk assessment should be recorded appropriately and used as the basis for making decisions about how to **manage** the risk. Finally, each risk assessment needs to be reviewed and updated periodically, and when new equipment or work practices are introduced.

15.2 HAZARDS AND RISKS

In general conversation, the words **hazard** and **risk** are often used interchangeably to denote a chance event whose outcome is undesirable, e.g. the risk of being struck by lightning. In formal risk assessments, however, it is useful to reserve the term **hazard** to describe any inherent property of an activity or situation that could potentially cause harm. Making this distinction, it follows that, just because something is hazardous, it does not mean that it will always result in a dangerous outcome – golfers will readily appreciate that hazards such as bunkers and trees are only dangerous if the ball is played into them!

As hazard is an inherent, dangerous characteristic of something, it is therefore a certainty. Risk, on the other hand, is a chance or, in mathematical language, a **probability**. Note that the value of any probability always lies between 1 (event certain to occur) and 0 (event certain not to occur). Risk can be defined as 'the probability that some harm will happen due to the realization of a hazard'. Sometimes risk is a single numerical probability but in most situations it is the product of a number of probabilities. It is very often expressed over a specific time period, e.g. a year, and then it becomes a frequency. Usually there is also an indication of how serious the harm could be, such as 'risk of death'. Table 15.1 compares the approximate values of a number of familiar risks.

Table 15.1 A comparison of some familiar fatal risks

Familiar risks	Risk of death (1 in *N*)
Smoking 10 cigarettes per day for a year	200
Being a deep-sea fisherman for a year	500
An effective (whole-body) radiation dose of 20 mSv (UK annual dose limit)*	1000
Radiation-induced fatal cancer from having one CT scan of the abdomen	2500
Being a coal miner for a year	6500
An accident at home over a 12-month period	10 000
Being murdered in a 12-month period	100 000
Being hit in your home by a crashing aeroplane in a 12-month period	250 000
Drowning in the bath in a 12-month period	685 000
Radiation-induced fatal cancer from having one chest X-ray examination	1 000 000
Being struck by lightning in a 12-month period	6 200 000
Death from a nuclear power station accident in a 12-month period	10 000 000

**It is common to express radiation risks as the probability of some adverse effect per unit dose. For example, the risk of fatal cancer is estimated to be about 5×10^{-2} per sievert effective dose, or 5 per cent per sievert. This means that if 1000 people each received an effective (whole-body) dose of 100 mSv then, on average, five of them would die of a radiation-induced cancer.*

Example 15.1

The risk of death from rock climbing is estimated to be about 4000 per 10^8 hours for the people involved. Calculate the annual risk of death for a climber who participates in the sport for 50 hours each year.

A risk rate of 4000 per 10^8 hours equates to 4000×10^{-8} or 4×10^{-5} per hour. The rock climber's annual risk of death is therefore

50 (hours per year) $\times$ 4×10^{-5} (risk per hour) = 2×10^{-3} or 1 in 500

15.3 THE BASIC STEPS IN RISK ASSESSMENT

The basis of all risk assessments is to ask a series of 'what if?' questions (e.g. 'What if the containment should fail?') and ensure that the answers are as comprehensive and accurate as possible. The answers are obtained by reference to practical experience of similar situations, experts' judgement, manufacturing standards and test results, and so on. The risk analyst then estimates the probabilities and consequences of all the foreseeable operating conditions that could arise, including non-standard operations, and makes an assessment of the overall risk. He then has to decide how to present the information so that it will be of maximum value in reaching decisions about the most appropriate control measures to be adopted.

Any risk assessment consists of a limited number of individual steps, as shown in Figure 15.1. These steps need to be carried out for any risk assessment, no matter how simple or complicated it is. However, it is important to ensure that each risk assessment is *suitable* and *sufficient* for the operation in question. Completeness is essential but the aim should be to achieve this without overcomplicating the risk assessment process unnecessarily.

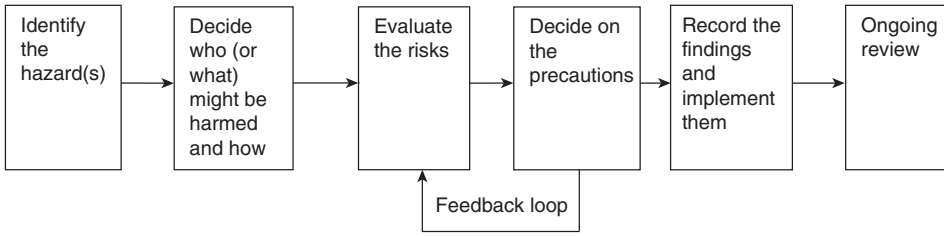


Figure 15.1 The basic steps in risk assessment.

Step 1: Identify the hazard(s)
There are various ways in which the hazards in a workplace can be identified. These include:

- physical inspection of the workplace to see what could reasonably be expected to cause harm;
- checking the manufacturer's instructions and data sheets to obtain information about the hazards associated with any new equipment or sources;
- carrying out a review of experience gained from similar operations;
- asking the operators – they may have practical insights not obvious to the risk assessor;
- reviewing the previous accident and ill-health records to see if there are any less obvious hazards.

All types of possible hazards should be considered, and one type, for example radiation, should not be solely concentrated on, to the exclusion of conventional industrial hazards.

Step 2: Decide who (or what) might be harmed and how
For each hazard, it is necessary to identify who might be harmed by it. For some hazards the potential damage may be to the natural environment rather than to people and this needs to be factored into the risk assessment. Having identified who or what might be harmed by the hazard it is then necessary to decide how they might be harmed, i.e. what type of injury, damage or ill health might occur. The following points are important:

- Different groups of workers, for example operators, contractors, maintenance staff and cleaners, are likely to be in the workplace for different periods of time and will therefore be exposed to the hazard in different ways.
- If members of the public could be exposed to the hazard this introduces additional complications in terms of the control and monitoring that can be applied to them. For example, the radiation risk factors for children are higher than for an adult population because (a) children are more likely to be alive for the whole latency period of the cancer, (b) children are more likely to have children of their own and (c) children's tissues are more radiosensitive than adults.
- There are particular requirements for the monitoring and protection of some workers, such as new workers, and new or expectant mothers, that need to be taken into account.
- If the hazard is likely to damage the natural environment it is necessary to determine the route(s) by which this might occur.

Step 3: Evaluate the risks

Having identified the hazard and who or what might be harmed by it, the next step is to evaluate the consequential risk, i.e. 'the probability that some harm will happen due to the realization of the hazard'. This is the core of the risk assessment process and relies on obtaining or generating a realistic estimation of the probability of harm. In some situations the probability of harm is quite obvious from a comparison with established good practice or industrial experience. However, in other situations, especially where the hazard is not immediately obvious to the normal human senses (e.g. ionizing radiation), the estimation of the probability of harm may be considerably more complicated. It usually involves estimating a number of probabilities and then multiplying them together, as shown in the following generic equation:

$$R = P(\text{haz}) \times P(\text{esc}) \times P(\text{int}) \times P(\text{dam})$$

where R is the risk; P(haz) is the probability that the hazard exists and is capable of causing harm to the subject of the risk assessment; P(esc) is the probability that the hazard 'escapes' from the devices incorporated to control it; P(int) is the probability that the hazard interacts in a damaging way with the subject of the risk assessment; and P(dam) is the probability that the interaction causes a specific type of damage in the subject of the risk assessment.

Every risk assessment should begin with an estimation of the harm that would result if the hazard were allowed to interact with the target (i.e. the person(s) or objects that might be harmed) without any specific precautions in place. This is sometimes known as the 'bare' risk assessment. In the 'bare' assessment, P(haz) and P(esc) are normally 1. P(int) is usually also 1, although there may be circumstances where it is less than 1. For example, members of the public would not normally suffer a damaging interaction with a sealed source housed in a controlled area unless there was an accident, such as a fire, that damaged the container of the sealed source and vaporized the radioactive material. In this situation, P(int) would be very small.

The final probability in the risk equation, P(dam), is relatively straightforward to assess when the potential harm is to the **safety** of the subject of the risk assessment, e.g. the probability that a dropped load which falls on a worker will result in a broken limb can be estimated from the weight of the load, the height of the fall and the likelihood of the worker being in the path of the falling load. However, in the case of ionizing radiation and toxic chemicals, where the potential harm is to the **health** of the subject, a whole range of possible health effects may need to be taken into account, both to the workers exposed in the workplace and to any members of the public exposed to releases from the workplace. For ionizing radiation, the probability and extent of such health effects will depend on the type and intensity of the radiation emitted, the age of the workers, whether any of them is pregnant, the composition and dietary habits of any groups of the public who may be exposed, the chemical form of any radioactive material released to the environment, and so on. Much of the information needed to form a reliable estimate of P(dam) is available in the recommendations and guides of the International Commission on Radiological Protection (ICRP), as well as in other national standards and guides, though additional information and tests may be needed in some situations, such as clean-up and decommissioning operations, in order to form a reliable estimate of P(dam). The normal approach for sealed sources is to estimate the unshielded dose rates (to the whole body and extremities) for the

proposed activity, multiply these by the time over which the activity is going to take place and compare the results with the relevant regulatory limits and local dose constraints. If the predicted dose exceed the limits or constraints then additional precautions, such as shielding and exposure time control, should be incorporated.

Having estimated the four probabilities it is then quite straightforward to arrive at the corresponding risk by multiplying them together. Note that it is often neither practical nor necessary to attempt to ascribe actual numerical values to the four probabilities in the risk equation. In many cases it is sufficient to decide that each of the probabilities is either **high**, **medium** or **low**. Table 15.2 illustrates, in broad terms, how the qualitative value of the risk might vary according to the qualitative values of the four probabilities.

Table 15.2 Illustration of qualitative risk assessment

***P*(haz)**	***P*(esc)**	***P*(int)**	***P*(dam)**	***R***
High	High	High	High	High or very high
Medium	High	High	High	High
Medium	High	Medium	High	High
Medium	Medium	Medium	High	Medium
Medium	Medium	Medium	Medium	Medium
Low	Medium	Medium	Medium	Medium
Low	Medium	Medium	Low	Medium
Low	Low	Medium	Low	Low
Low	Low	Low	Low	Low or very low

When carrying out an actual risk assessment, however, it is always necessary to look carefully at the importance of each of the probabilities in the context of the hazard being controlled rather than following this table 'blindly'.

Example 15.2

Calculate the risk that a cyclist will be seriously injured by a car at a main road junction based on the following information:

Probability that a car is approaching the main road as cyclist passes the junction: 0.01

Probability that the car fails to stop (due to driver error or inattention, road conditions, equipment failure, etc.): 0.05

Probability that the car hits the cyclist (either the driver or the cyclist may take successful evasive action): 0.5

Probability that the impact causes serious injury to the cyclist (likely to be less than 1 if cyclist is wearing a protective helmet): 0.8

Using the equation $R = P(\text{haz}) \times P(\text{esc}) \times P(\text{int}) \times P(\text{dam})$

The risk in this case is $R = 0.01 \times 0.05 \times 0.5 \times 0.8 = 2 \times 10^{-4}$

This means that, under these assumptions, if 10 000 cyclists use the main road every year, it is expected that about two of them will be seriously injured in a crash at this junction.

Step 4: Decide on the precautions

Having estimated the 'bare' risks associated with a particular work activity it is then necessary to decide on the precautions required. In some circumstances, there will be national guides or regulatory requirements that define the risk level at which specific precautions must be introduced. These may also define the type of precautions that should be taken. In other situations, there may simply be the general legal requirement to do everything 'reasonably practicable' to protect people and the environment from harm or to ensure that doses are as low as reasonably achievable (ALARA). For some work activities, what is 'reasonably practicable' can be deduced from a comparison with accepted good practice. Where this is not possible, the responsible employer may have to work out what is 'reasonably practicable' for themselves, following the rule that precautions to eliminate or reduce the risk should be incorporated until their cost becomes quite disproportionate to the value of the risk being averted.

When controlling risks, the following principles should be applied, in the order shown where possible:

- consider a less risky option (e.g. use a smaller radioactive source or a less hazardous chemical);
- prevent access to the hazard (e.g. by using a ventilated glove box);
- organize the work to limit exposure to the hazard (e.g. establish contamination control areas and barrier controls);
- define and issue appropriate protective equipment (e.g. clothing, footwear, masks, goggles);
- incorporate the above in appropriate operating procedures and provide training as necessary;
- provide welfare facilities (e.g. first aid and washing facilities for the removal of contamination); and
- if indicated by the risk assessment, provide facilities and resources to deal with potential accident situations.

Having decided on the precautions it is then necessary to follow the feedback loop shown in Figure 15.1 and redo the risk estimation with the precautions included. This will alter some or all of the probabilities in the risk equation and result in a new estimate of the risk. This is again compared against the target for reasonable practicability to decide if it is acceptable. If it is still unacceptable, further precautions are included and the process repeated until the situation is judged to be satisfactory. Note that the introduction of a precaution or control can increase the risk of injury from some other hazard. For example, medical X-ray staff wear lead-impregnated aprons that are 0.25 mm lead-equivalent to protect them from scattered radiation (attenuation factor around 90 per cent). Higher lead equivalences (e.g. 1 mm) would obviously give greater attenuation but the risk of back injury from the weight of the apron begins to outweigh the increased radiation shielding benefit.

Step 5: Record the findings and implement them

It is always good practice to record the findings of each risk assessment. The written record provides the basis for improving the work activity and allows the results to be shared with the staff who will be involved, as well as with other people who might be affected by the

work activity. It allows appropriate operating rules to be developed and assists in defining training requirements.

Step 6: Ongoing review
It is essential to ensure that the risk assessment for any work activity is up to date. When any changes are made, such as bringing in new equipment, substances or procedures, the risk assessment should be reviewed. It should also be reviewed on a regular basis to see if there are any improvements outstanding from previous reviews, if the operators have identified any problems 'on the ground', or if there are lessons to be learned from accidents or near misses.

15.4 HAZARD AND RISK IN RADIATION PROTECTION

Radiation protection is essentially concerned with controlling the risks that arise from the use of radioactive materials or machines that produce ionizing radiations. The nature of ionizing radiation means that it is normally impossible to eliminate all risks from activities involving these sources of hazard. In a facility that has been designed, maintained and operated properly, the risk should be very small, but it is not zero. In any situation, the risk is estimated by identifying all the possible chance events (such as material leaking from a radioactive source or a radiation machine malfunctioning; a worker or member of the public being in the 'path' of the escaping radiation, etc.) that could lead to the ionizing radiation reaching and interacting with its 'targets'. Then a probability is ascribed to each of these chance events happening, either from experience or from experiments, measurements or modelling, and the risk is calculated by multiplying all these probabilities together, as shown by the previous generic risk equation. This is a simplified description of the process of risk assessment, which can often turn out to be quite complicated in practice. The following two examples illustrate how the risk is assessed in two quite different situations, following the six steps in Figure 15.1.

Example 15.3: risk assessment prior to commencing work with a sealed source in a radiochemistry laboratory

Regulation 7 of the UK Ionizing Radiations Regulations (1999) requires that:

'before a radiation employer commences a new activity involving work with ionizing radiation in respect of which no risk assessment has been made by him, he shall make a suitable and sufficient assessment of the risk to any employee and other person for the purpose of identifying the measures he needs to take to restrict the exposure of that employee or other person to ionizing radiation.'

Step 1: Identify the hazard
Although the general hazard is clearly ionizing radiations from the sealed source, the risk assessor needs to know the specific radioisotopes present, their activity levels and principal emissions, and the energies of the principal emissions. This information will be required later to allow the doses to various subjects to be estimated.

Step 2: Decide who (or what) might be harmed and how
It is likely that the only persons who might be harmed during normal operations would be the laboratory staff working with the sealed source. However, the possibility that other workers, such as cleaners or maintenance staff, might be harmed needs to be considered, as well as the possibility that workers and members of the public might be harmed in abnormal operating conditions or as the result of an accident, such as a fire.

Step 3: Evaluate the risk
As discussed in section 15.2, the first step is to estimate the 'bare' risk from the sealed source by calculating the unshielded dose rates for the activity in the sealed source. This would be carried out for appropriate distances from the source and compared against regulatory targets and local dose constraints. The possibility of workers receiving an intake of radioactivity due to a leakage from the source would be assessed by taking wipe tests of its surface. This would allow the magnitude of any such intake to be estimated. Finally, the risk from potential accidents, such as a fire in the laboratory, needs to be estimated.

Step 4: Decide on the precautions
Depending on the results of the 'bare' risk assessment, various precautions (e.g. local shielding, operating time restrictions, face masks, fire sprinklers or air filters) might be incorporated. The risk estimation would then be repeated with all the precautions in place to see if the resultant doses are ALARA. If necessary, additional precautions might be incorporated at this stage and the process repeated until a satisfactory outcome is achieved.

Step 5: Record the findings and implement them
In most laboratories there is a standard form for recording the findings of the prior risk assessment and the precautions to be taken. This has to be completed and signed off by the responsible radiation protection officer before any work is started.

An example of such a form is shown in Table 15.3. It covers the risk assessment of a hand-held dental X-ray machine in a teaching laboratory. Note that this is an example and is not intended to give a complete coverage of all the information that would normally be included in such a form.

Comparing this form with the six basic steps in the risk assessment process, it can be seen that:

- column 1 corresponds to step 1 (identify the hazard);
- column 2 corresponds to step 2 (decide who or what might be harmed and how);
- columns 3, 4 and 5 correspond to the feedback loop shown in Figure 15.1 (evaluate the 'bare' risks, decide on the precautions, re-evaluate the risks with the precautions included, assess the resultant risks and decide if any further actions are needed).

The completed form corresponds to step 5 (record the findings and implement them) and it should contain a requirement for the risk assessment to be reviewed periodically or when alterations are made or there has been an incident (step 6). Finally, the form should be dated and signed by the person responsible for carrying out the risk assessment once all outstanding actions have been completed.

Table 15.3: Risk assessment for a hand-held X-ray machine

Hazard description	Who might be harmed?	Existing control description	Risk (S, M, L)	Comments or any further actions required to reduce the risks
External exposure of staff to X-rays in routine practice	Dentists, dental nurses or technicians	Local rules available. Staff must read these, then sign and date a register to verify understanding. The controlled area is nominally within 1 metre of the head phantom and anywhere in the primary beam. The X-ray machine must be operated only with a Perspex shield fitted to the end of the collimator. During exposures staff will always stand outside the controlled area as defined in the local rules. Lead aprons available if needed	S	Personal monitoring is felt unnecessary due to low workload and adherence to working procedures. A warning sign must be displayed outside each door during X-ray procedures. **No further action is needed**
External X-ray exposure to persons outside the room in routine practice	Patients or staff in adjoining rooms or outside. General public	The exterior walls of all surgeries are constructed of solid brick. The primary beam is only ever directed towards a solid brick wall from over 1 metre away. It is never directed towards any door, the ceiling or a stud wall	S	Primary beam output = 86 μSv/h at 1 metre. Normal brick density = 1.8 g/cm^3. Linear attenuation coefficient at 60 kV is 0.48 cm^{-1}. Instantaneous dose rate outside room is < 2.5 μSv/h. Combining output and workload of X-ray machine with occupancy of adjacent rooms, the relevant dose constraint is unlikely to be exceeded. **No further action is needed**
External exposure of staff or patients in abnormal or fault conditions	Patients, dentists, dental nurses or technicians	No patients X-rayed, machine used only for student practicals. Local rules contain contingency plan in case machine fails to terminate. Machine serviced and maintained annually	S	**No further action is needed**

Example 15.4: Risk to personnel during decommissioning

In the above example the hazard of greatest concern is external radiation. The potential for significant radioactive contamination is limited to a few, highly unlikely, accident scenarios. The assessment of conventional hazards such as slips, trips and falls is relatively straightforward for such dedicated, well-controlled facilities. In contrast, the risk assessment of decommissioning activities is often much more complicated because of the presence of both external and internal radiation hazards and the requirement for personnel to work in obsolete facilities where many conventional hazards may be present. This is illustrated by the following example, which involves the decommissioning of a facility containing a number of remote-handling cells that had been used over several decades for handling a variety of radioactive materials, including uranium and plutonium. The steps in the risk assessment are as follows.

Step 1: Identify the hazard

A variety of hazards may need to be taken into account in the risk assessment, such as:

1) ionization radiations emitted from both fixed and loose radioactive materials
2) uncontained radioactive gases, aerosols and liquids
3) other hazardous chemicals, such as sodium and mercury
4) contact with machinery or plant
5) entanglement in machinery or equipment
6) electrocution
7) pressure/stored energy
8) working in confined spaces, e.g. flammable/explosive/toxic atmosphere, excessive heat
9) unstable structures
10) falling objects or materials
11) working at height/falls
12) slips and trips
13) lone working/becoming trapped
14) manual handling
15) accidents such as fire or explosion
16) release of radioactive or toxic material to the natural environment
17) radioactive and toxic material, e.g. asbestos, waste.

Step 2: Decide who (or what) might be harmed and how

The first 15 of the identified hazards would affect personnel working in the facility. The risk analyst has to consider the various subgroups involved (such as mechanical and electrical fitters, health physics monitors, supervisors, process workers, emergency response teams, safety inspectors) and decide which hazards are relevant to each subgroup. A risk analysis is then generally carried out for each subgroup. Other site personnel and members of the public, both on and off site, as well as the natural environment, could potentially be affected by hazards 2, 15, 16 and 17. Appropriate risk assessments would be necessary for each of these.

Step 3: Evaluate the risk

The first step in the process is to evaluate the 'bare' risk, i.e. the risk with no controls in place. This involves a combination of the radiation-related risk and the conventional, industrial risk. The risks associated with potential exposures to ionizing radiation are considered first using the risk equation:

$$R = P(\text{haz}) \times P(\text{esc}) \times P(\text{int}) \times P(\text{dam})$$

***P*(haz)**: In this decommissioning situation it is likely that there will be a number of well-recorded radiation and contamination sources as well as some others which either are suspected to be present or could be created by the decommissioning activities themselves. The risk analyst should consult all the relevant drawings and records, review operational experience, carry out calculations, and so on, in order to evaluate the probability that each hazard exists. For well-established hazards the probability will be 1; for suspected but unconfirmed hazards it may be around 0.5; while for hazards that are unlikely to exist but cannot be ruled out entirely, the probability may be 0.1 or less.

***P*(esc)**: In order to assign a value of *P*(esc) to each hazard, the risk analyst needs to evaluate the physical and chemical properties of the radioactive materials within the facility and the operations that will be carried out during the decommissioning. For example, it may be necessary to estimate the likelihood that cutting and grinding operations will lead to the creation of airborne radioactive dust.

***P*(int)**: For some decommissioning activities *P*(int) may be less than 1, if the radiation source is fixed and cannot be disturbed by the decommissioning activities. Generally, however, the nature of the decommissioning process means that *P*(int) is 1 unless specific measures, such as protective clothing, face masks or breathing apparatus, are incorporated to protect the workers.

***P*(dam)**: The probability that the interaction of the emitted radiations will cause a specific type of damage (health effect) in the exposed workers will take various values depending on such things as the expected mobility and physical/chemical properties of the radioactive materials, the type and energy of the emitted radiations and the age and sex of the workers. The information needed to estimate *P*(dam) is usually available in the recommendations and guides of the ICRP or in national guidance.

The 'bare' risk to each of the different categories of workers, members of the public and the environment from potential radiation exposures arising from each radiation hazard is then estimated by multiplying the four probabilities together. The magnitude of each of these risks, in association with the predicted doses, indicates the need for protective measures to be incorporated to make the doses ALARA.

The conventional risk must also be considered, though it is likely that a qualitative approach, such as that illustrated below, will be adequate. Under this approach, the matrix shown in Table 15.4 would be constructed to give a qualitative estimate of the risk.

Table 15.4 Qualitative risk matrix

		Likelihood		
		Low (L)	**Medium (M)**	**High (H)**
Severity	**Low (L)**	Insignificant risk	Low risk	Medium risk
	Medium (M)	Low risk	Medium risk	Medium risk
	High (H)	Medium risk	High risk	High/very high risk

For **likelihood**, H (high) means that an injury is likely to occur; M (medium) means that it could occur; and L (low) means that it is unlikely to occur. For **severity**, H (high) means that the type of injury is likely to be death, permanent disability or long-term ill health; M (medium) means that broken limbs or injuries requiring hospital treatment are likely; and L (low) means that only minor injuries, such as bumps, bruises or cuts, are likely. The risks associated with each of the conventional hazards identified in step 1 would then be estimated and appropriate control measures incorporated. Note that, while the radiation and conventional risk assessments are often carried out separately, their predictions need to be coordinated in order to look for potential interactions that might alter one or more of the probabilities, or impact on some of the protective measures proposed.

Step 4: Decide on the precautions

As noted in section 15.3, the following principles should be applied when controlling risks:

- Consider a less risky option. For this major decommissioning exercise, this would imply a comprehensive review of the various possible options for carrying out the project. The review would look at such things as the balance between manual and automated procedures, the amount of pre-cleaning carried out and the methods to be used, the number of staff and the time required for different options, etc. The aim should be to arrive at a solution that achieves the optimum balance between the various risks (i.e. radiation, conventional, chemical, etc.) and which is expected to result in the lowest overall risk that is reasonably practicable. Note that this optimum balance may need to be amended or refined when the later principles are applied.
- Prevent access to the hazard. In order to carry out a decommissioning project, it is necessary for the workers to have access to the facility with its various hazards. However, there may be some localized areas of high hazard that can be cordoned off or shielded while the workers are decommissioning the remainder of the facility. All such opportunities for preventing access to the various hazards should be explored.
- Organize the work to limit exposure to the hazard. This will involve establishing contamination control areas and barrier controls, as well as optimizing the working arrangements for the various groups of staff involved.
- Define and issue appropriate protective equipment. The requirements for special clothing, footwear, masks, goggles and so on should be based on the best information available about the gaseous and airborne particulate activities likely to be present. They should also take account of the environmental conditions in the facility (e.g. state of the floor, ambient temperature) and the requirement to minimize the doses received by workers carrying out the decommissioning. For example, any requirement for fully enclosed, air-fed suits needs to be considered carefully in view of the loss of flexibility and manual dexterity, as well as the increased time to complete the task, that such use generally entails. It is also vital to ensure that the facial protection specified provides the appropriate level of protection against the gaseous and airborne hazards. Some face masks, for example, provide protection against airborne particulate activity but not against gaseous activity. Overall, the specification of appropriate protective equipment requires a careful and skilful balancing of

the various hazards likely to be present during such a major decommissioning exercise and it needs to be kept under continual review as the work progresses.

- Incorporate the above in appropriate operating procedures and provide training as necessary. The procedures for carrying out any decommissioning activity should reflect the provisions that have been incorporated to control the risks and be fully documented. For some particularly complicated decommissioning exercises, it may be appropriate to use non-active mock-ups of the facility for training staff.
- Provide welfare facilities. The extent and sophistication of the first aid and washing facilities for the removal of contamination should reflect the nature and magnitude of the hazards. They should always be as near as practicable to the facility being decommissioned.
- Provide facilities and resources to deal with potential accidents. The likelihood and severity of any potential accidents that might occur during the decommissioning process, as revealed by the risk assessment process, will dictate the scope and type of emergency arrangements needed.

Step 5: Record the findings and implement them.
The findings of the risk assessment need to be recorded in writing in order to set down the initial assumptions, which may need to be altered as experience and further information are gained from the decommissioning activities, and to facilitate the production of appropriate operating procedures and instructions. In some circumstances, a written risk assessment may be a requirement for obtaining regulatory approval to carry out the decommissioning.

Step 6: Ongoing review
The risk assessment for a major decommissioning exercise such as this should be reviewed on a continuous basis. As decommissioning progresses, hazardous equipment and materials will be removed but new ones may be created by the decommissioning process itself. Hidden radioactive substances may be revealed that require different control measures. The composition of the workforce or the working environment may change. Therefore, the risk assessment should be looked at constantly with the following questions in mind: Have there been any changes? Are there improvements still to be made? Have the workers spotted any problems? Are there any lessons to be learned from accidents or near misses?

15.5 PROBABILISTIC RISK ASSESSMENT

In the previous examples much of the risk assessments would generally be carried out in a qualitative way, although it would normally be necessary to calculate the doses that the subjects at risk would experience as a consequence of the hazards. However, for some purposes, particularly the assessment of the risk from remote, high consequence accidents on complex plants such as aircraft or nuclear reactors, it is vital to use a quantitative risk assessment technique to ensure that all the possible paths to an accident are identified and their probabilities and consequences evaluated fully. One of the most important

approaches is known as **probabilistic risk assessment** (PRA), also sometimes called **probabilistic safety assessment** (PSA).

There are two fundamental quantities in a PRA:

a) the magnitude or severity of the possible adverse consequences; and

b) the likelihood or probability of occurrence of each consequence.

The consequences are expressed numerically (e.g. the number of people potentially injured or killed, the area of land contaminated) and their likelihoods of occurrence are expressed as probabilities or frequencies (i.e. the number of occurrences or the probability of occurrence per unit time). The total risk is the sum of the products of the consequences multiplied by their probabilities.

PRAs are normally developed using **fault tree analysis** or **event tree analysis**. Sometimes a combination of the two is used. In a fault tree analysis initiating primary events are traced through a tree logic system to an undesired top event, such as a reactor core melt. Fault trees consist essentially of a series of 'and' and 'or' gates. Figure 15.2 illustrates a simple fault tree, for a fire breaking out. This is, of course, only a small part of the fault tree that would be needed to identify the root cause of this event. Note that, at an 'and' gate, both contributors are required for the event to occur and so the probabilities are multiplied. In contrast, at an 'or' gate any one of the contributors is sufficient for the event to occur and so the probabilities are added.

In contrast to fault tree analysis, event tree analysis starts from an undesired initiator, such as loss of electrical supply or component failure, and follows possible further system events through to a series of ultimate consequences. At the point where each new event arises, a new node is added to the tree with a split of probabilities for each new branch. The probabilities of a range of top events arising from the initial event can then be determined. Figure 15.3 illustrates a simple event tree.

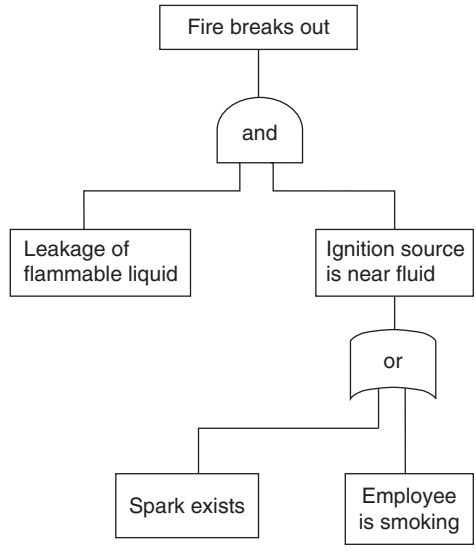


Figure 15.2 Fault tree for a fire breaking out.

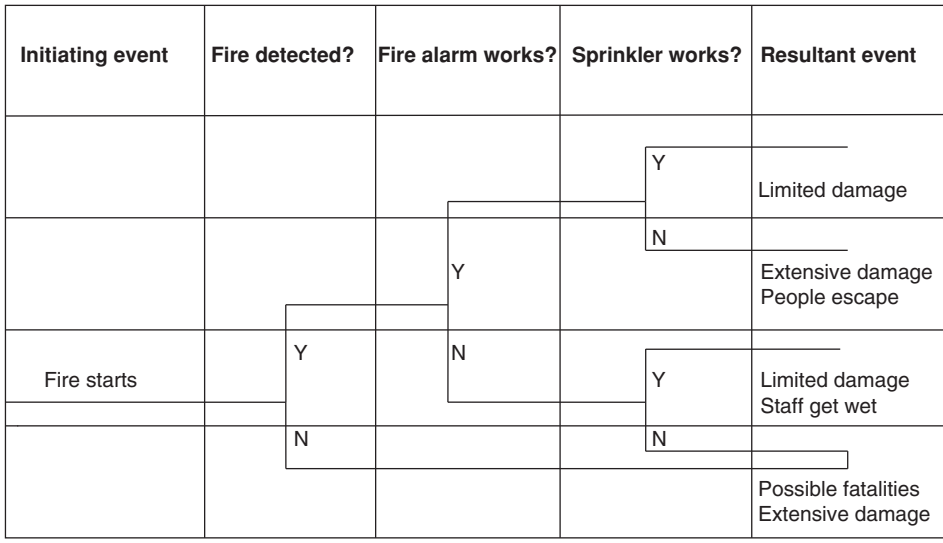


Figure 15.3 Event tree for a fire in an office building.

15.6 UNCERTAINTY, SENSITIVITY AND ACCEPTABILITY

In order for a comprehensive risk assessment to be useful in decision-making, it needs to include all the relevant risks and give an accurate estimate of their magnitude. However, no matter how precise any risk estimation appears to be, it is always potentially subject to some ignorance about the hazards present as well as to a number of inescapable uncertainties. For example, in the decommissioning example considered earlier, it may be impossible to know for certain the complete range of radioactive substances present. This lack of knowledge can be countered by involving the widest possible range of expertise and experience in the hazard identification process, but it is often not entirely removable.

The uncertainties in the risk assessment process can be grouped under the two general headings:

a) modelling uncertainty, where there is a lack of certainty about the validity of the model used to represent the process giving rise to the risks; and

b) knowledge uncertainty, when the data in a risk assessment are based on sparse statistics or are subject to random experimental errors.

The effects of these uncertainties are studied by means of sensitivity analyses in which the data and modelling assumptions are changed systematically, within the range of their uncertainties, to assess their effect on the final outcome. This provides an estimate of the reliability of the risk calculation as well as identifying the importance of different sources of uncertainty.

Having calculated the risk and established how reliable the answer is, the decision-maker then has to judge whether or not it is acceptable to undertake the activity. For activities involving ionizing radiations, the ICRP's recommendations can be seen as representing the best current international consensus on risk acceptability. Thus, its recommended annual whole-body dose limit of 20 mSv corresponds to a risk of fatal cancer induction of

approximately 1 in 1000, which may be regarded as just tolerable. The risk can be scaled up and down linearly depending on the dose so that, for example, a calculated annual dose of 40 mSv gives a risk of 1 in 500, which is unacceptable, while a calculated annual dose of 1 mSv gives a risk of 1 in 20 000, which is acceptable. In other situations what is acceptable may be judged on the basis of accepted current best practice or the use of 'state-of-the-art' technology.

The most difficult questions of acceptability are associated with the risks from very unlikely but high consequence accidents that have the potential to affect large portions of the country and society, as well as the local workers and members of the public. A number of countries have developed acceptability criteria for such risks. For example, in the USA the Nuclear Regulatory Commission has issued a policy (in 10 CFR Part 50) that establishes safety goals that broadly define an acceptable level of radiation risk in terms of two qualitative and two quantitative goals linked to additional prompt fatalities and additional cancer fatalities. The UK's acceptability criteria are based on a framework, known as the tolerability of risk (TOR), which was developed to meet a recommendation of the public inquiry into the Sizewell B nuclear power station and first published in 1988.More detailed information on modelling uncertainty, knowledge uncertainty, sensitivity analysis, acceptability and TOR is provided at http://www.hse.gov.uk/nuclear/tolerability.pdf

SUMMARY OF KEY POINTS

Hazard: an inherent property of an activity or situation that could potentially cause harm.

Risk: the probability that some harm will happen due to the realization of a hazard.

Risk assessment: the process of identifying the hazard, estimating the size of the risk and assessing its importance in comparison with other risks.

Steps in risk assessment process: identify the hazard(s), decide who (or what) might be harmed and how, evaluate the risks, decide on the precautions, record the findings and implement them, and carry out an ongoing review.

Probabilistic risk assessment: a quantitative, systematic and comprehensive methodology for evaluating the risks associated with complex engineering plants such as nuclear power stations.

Fault tree analysis: a method in which initiating primary events, such as component failure, are traced through a tree logic system to an undesired top event, such as a reactor core melt. It is a valuable technique for identifying the failures that have the greatest influence on bringing about the top event.

Event tree analysis: starts from an undesired initiator, such as a loss of electrical supply, and follows further system events through to a series of ultimate consequences. Valuable in analysing the consequences resulting from a failure or undesired event.

Uncertainty in risk assessment: consists of two parts, modelling uncertainty and knowledge uncertainty.

- **Modelling uncertainty:** relates to the validity of the model used in the risk assessment process.
- **Knowledge uncertainty:** relates to uncertainties in the data used in the model.

Sensitivity analysis: the process by which the data and modelling assumptions are changed systematically to assess their effect on the final outcome.

Acceptability: the process by which the decision-maker decides if the outcome of the risk assessment is acceptable in comparison with other risks.

REVISION QUESTIONS

1. Explain the difference between hazard and risk.
2. If the risk of death from accidents in the home is 4 per 10^8 hours, calculate the approximate number of deaths that might be expected to occur over a year in a population of 100 000 who each spend 100 hours per week at home on average.
3. Describe the six basic steps in risk assessment and explain how you would apply them in assessing the risk associated with carrying out a dental X-ray.
4. Explain the two fundamental quantities involved in a probabilistic risk assessment.
5. Explain the difference between a fault tree and an event tree.
6. Draw a simple event tree and calculate the probability of a release to atmosphere of fission products as the result of a fuel element leak in a pressurized water reactor (PWR) using the following information: probability of fission products being carried by coolant to the steam generator, 0.9; probability of leak in steam generator tubes, 0.01; probability of fission products being ejected from the steam generator condenser, 0.9; probability of fission products escaping from the containment building, 0.1.
7. Explain the difference between modelling and knowledge uncertainties in risk assessment.

16 Radiological incidents and emergencies

16.1 INTRODUCTION

A radiological incident may be defined as any unplanned situation that gives rise to an abnormal radiation hazard. This definition covers anything from a minor laboratory spill involving a few megabecquerels (MBq) of radioactive solution up to a major reactor accident in which many thousands of terabecquerels of fission products may be released into the environment.

An incident can arise because of:

1. a **loss of shielding**, resulting in high radiation levels;
2. a **loss of containment**, resulting in a release of radioactivity; and
3. an **uncontrolled criticality**, which is, effectively, the rapid generation of a large radioactive source and high levels of radiation.

Usually, these situations result from some conventional cause such as a mechanical failure, fire, flooding or a transport accident.

For planning and control purposes, it is usual to differentiate between the various levels of events that can occur. Minor events such as the laboratory spills mentioned above are more of a nuisance than a danger and it is more appropriate to refer to them as **local incidents**. The effects would normally be confined to a very limited area and would be controlled by simple measures such as cleaning up the spill. A more serious situation with the potential to give rise to significant radiation doses to personnel on the site but with no effect outside the site or establishment could lead to the declaration of a **site emergency** and the initiation of pre-planned procedures to protect personnel and to bring the situation under control. At a still larger scale, if the incident could have effects outside the site, it would be declared a **public emergency**, again requiring implementation of an emergency plan. The International Atomic Energy Agency has devised an International Nuclear and Radiological Event Scale (INES) to be applied following any unplanned occurrence that has a potential radiological impact (see section 16.2).

Whatever the scale of the event, it is essential to have analysed the possible occurrences in advance and to have formulated plans and procedures for dealing with them. It is vitally important to detect any abnormal situation as quickly as possible. For example, if a loss of shielding accident is detected immediately and the appropriate corrective actions or evacuation measures are taken, the dose received may be very small. Conversely, very large doses may be received if operating personnel and others in the vicinity of the plant are not aware of the situation. In this chapter, a number of situations of the types listed above will be considered.

16.2 INTERNATIONAL NUCLEAR AND RADIOLOGICAL EVENT SCALE

INES was devised by an international group of experts convened by the IAEA in 1989, largely as a result of lessons learned from managing the Chernobyl accident (see section 16.4.3). The scale uses a numerical rating to indicate to the wider community, both nationally and internationally, the magnitude of an event in a similar way to that used to indicate the magnitude of an earthquake on the Richter scale. Events are classified at seven levels, with Levels 1–3 considered to be incidents and Levels 4–7 considered to be accidents. The levels are broadly defined as follows:

Level	Type of event
1	Operating anomaly with minor impact
2	Incident leading to some over exposure to radiation
3	Serious incident, with actual or potential high exposures
4	Accident with local consequences and minor release of radioactive material
5	Accident with wider consequences and likely to require countermeasures
6	Serious accident with significant release of radioactive material requiring countermeasures
7	Major accident with widespread health and environmental effects

Examples are given in the following sections of the application of INES. However, it should be noted that whilst the system was developed internationally, it is applied at a local or national level, and this has sometimes led to inconsistencies in the rating of events.

16.3 LOSS OF SHIELDING

16.3.1 Small sealed sources

Small sealed sources, usually gamma (γ) emitters, are widely used in industry, medicine and teaching. It is unlikely that any source with an activity less than 100 MBq could result in an excessive dose to a person (unless, for example, it was carried in the pocket) and so the loss of shielding for such sources would probably be a local incident. Such sources are usually handled by tongs and stored in small lead-lined pots. The most common loss of shielding incidents occur when a source is removed from its container by some person who fails to return it after use. The best method of protecting against such incidents is to use installed monitoring and/or alarm instruments that show when the source is out of its pot. However, in the majority of cases, reliance is placed on portable monitoring equipment. Regular source musters minimize the possibility that the situation remains undetected for a long period of time.

The loss of shielding could also result from mechanical damage, for example if the container were dropped, in which case there should be no problem in detecting the event. The possible effects of fires, which might not only cause the shielding to melt but also cause a loss of containment of the source material itself, must also be considered. Events involving small sealed sources would normally be classed as Level 2 on INES.

16.3.2 Large sealed sources

Large sealed sources such as those used for industrial processing or radiography and medical radiotherapy are usually housed in specially constructed containers with mechanical means for controlling the time of exposure. The containers are designed to withstand foreseeable mechanical accidents and to resist fire. The possibility of inadvertent exposure of the source is minimized by the design of the equipment, but alarm systems are desirable for detecting any fault conditions.

The majority of accidents involving sources of this type have been in industrial radiography. This is often performed in difficult conditions on construction sites without any form of installed monitoring equipment. In a number of accidents the radioactive source became detached from the operating mechanism and, when the mechanism was retracted into the storage position, the source remained unshielded. In some cases, the source was found by a person who placed it in their pocket without being aware of its hazardous properties. This resulted in very large, and sometimes fatal, doses being received. The prevention of such accidents depends not only on the correct use of appropriate equipment but also on good training and strict adherence to pre-planned monitoring procedures by the radiographer. Various portable alarm devices are also available which can be used in field conditions. There should also be a contingency plan to deal with accidents and occurrences, and equipment should be readily available to assist in recovery of the situation.

Past events involving large sources have been classified between Levels 2 and 5 on INES, depending on the circumstances and the impact.

16.3.3 Entry into shielded cells

Another situation with the potential to give rise to very serious exposures is accidental entry into a shielded cell containing a large γ source or equipment for generating high radiation fields, such as X-ray machines or linear accelerators. The consequences of such entry can be so severe that it is essential that safety is a prime factor in the design of the facility. Interlocked systems should prevent the equipment being activated when an access door or port is open and, conversely, prevent a door being opened when the equipment is operating. In addition, installed monitoring and alarm equipment should be incorporated into the design. It is also important that safety features should be designed to ensure that the facility fails in a safe mode. Strict adherence to the entry and portable monitoring procedures provides a further layer of protection.

An accident occurred in 1991 at a Teflon treatment facility in Forbach (France) where an electron accelerator irradiator was being used to treat materials. In order to save time, three workers had entered the irradiation room via an exit. Although the accelerator was switched off, the accelerating voltage was not (known as 'dark current' mode) and the dose rate in the room still ranged from 100 mGy/s up to several grays per second (as opposed to 80 000 Gy/s when the accelerator was on). The three received localized doses, one severe enough to produce skin lesions. The skin doses were estimated at 40 Sv (effective dose of 1 Sv) for the worker with the worst injury, and 9 and 5 Sv for the other two workers.

16.3.4 Reactor fuel-handling accidents

The problems and dangers involved in handling the intensely radioactive fuel from a nuclear reactor are described in section 11.4. For large power-generating reactors, complex,

remotely controlled handling equipment removes fuel from the reactor and transfers it to a cooling pond. It is virtually impossible, because of built-in safety devices, for fuel to become unshielded at this stage. With research reactors, there is usually more scope for fuel-handling accidents and greater reliance has to be placed on following approved operating procedures. There is greater scope for loss of shielding accidents in fuel cooling ponds, perhaps through the inadvertent withdrawal of fuel from the pond or because of loss of water. The possibility of such occurrences is minimized by good equipment design and careful operation and maintenance but, as a final safeguard, an installed radiation alarm system is essential.

Accidents of this type are rare but they would generally be classified as Levels 2 or 3 on INES.

16.4 LOSS OF CONTAINMENT

16.4.1 Minor spillage of radioactivity

Perhaps the most common 'abnormal occurrence' in a laboratory is a minor spillage of up to a few megabecquerels of radioactive solution. The frequency of such events is minimized by good laboratory practices such as keeping containers of radioactive solutions in trays to contain any spillage. Spills do, however, occur even in the best-regulated laboratories but, if they are dealt with correctly, the contamination and therefore the incident should not spread outside the laboratory or area in which it occurred. After carrying out any personnel decontamination that may be required, the most important action is to clean up the radioactive contamination using absorbent materials before it dries out and becomes airborne.

A useful precaution in laboratories handling unsealed sources is to have available a few spillpacks. A spillpack is simply a plastic bag containing a pair of gloves, a pair of overshoes and a wad of absorbent material (cotton waste, paper towels, etc.). When a spill occurs, the gloves and overshoes are donned and the spill is wiped up using the absorbent material, which is then replaced in the bag for disposal. Having quickly cleaned up the bulk of the activity, the surface can then be monitored and decontaminated further if necessary. If any person is contaminated as a result of the spill, they should put on the clean gloves and overshoes and go to a change or decontamination area without spreading contamination.

Spillages and leakages can also occur in more industrial environments, including nuclear facilities, where the nature of equipment and surface finishes in the area are less amenable to decontamination. Whatever the environment, any spillage that remains undetected for some time is likely to cause problems because it will be spread around, possibly outside the area. This is why it is important that areas are subject to routine monitoring and that personnel leaving the area follow strict washing and monitoring procedures.

A release of radioactivity can also result from a failure of services such as ventilation or electrical supplies. Glove boxes can pose a particular problem in this respect. A glove box is normally operated at a pressure slightly below atmospheric, which means that leakage tends to be in rather than out. If some failure causes the box to pressurize, outward leakage may occur or, more seriously, a glove or panel may be blown out resulting in a release. Specific attention needs to be paid to such possibilities in the design in order to minimize their chances of occurring, and there should be pre-planned procedures for dealing with them.

Events of this type would normally be considered as minor incidents and hence be rated as Level 1 on the INES system.

16.4.2 Major spills of radioactivity

A major spill, involving more than 100 MBq or so of activity, depending on the radiotoxicity of the nuclides involved, could result in a significant incident. The INES rating could be Level 2 or 3. Immediate evacuation of personnel might be required together with shut-down of the ventilation system and sealing the area off to contain the spread of activity. A controlled re-entry to the area by a team wearing appropriate protective clothing and respirators might be necessary. It is circumstances such as these that demonstrate the value of a properly designed facility. Decontamination can be a relatively simple matter where proper attention has been paid to surface finishes. In badly designed facilities, decontamination may be difficult or even impossible.

16.4.3 Major releases from nuclear facilities

Potentially the most serious loss of containment accidents involves the release of fission products from a reactor. It will be recalled that the fission products are contained within three separate boundaries: the fuel cladding, the boundary of the cooling system and the reactor building. In a power reactor, the most likely cause of a fission product release is a loss of cooling with subsequent overheating and, possibly, melting of the fuel. Loss of cooling could occur because of a failure of the pressure circuit and consequential loss of coolant or because of failure of pumps or other equipment on which the cooling of the core depends. Fission products would then be released from the molten fuel and escape through any breach in the cooling system. In a large reactor, if only 0.1 per cent of the fission product inventory leaked from the cooling system, this could amount to over 10^{17} Bq. If 1 per cent of this amount then escaped from the reactor building or containment, the release to the environment would be 10^{15} Bq. This would result in very high levels of radiation and contamination on the reactor site. It would also be a hazard to the local population and hence a public emergency.

It was considerations such as these that led to the siting of the first generation of nuclear reactors in remote areas. Some large plants of more recent design have been sited quite close to towns. This is justified either because, in plants such as the advanced gas-cooled reactor (AGR), the entire reactor system is inside a massive pre-stressed concrete pressure vessel and the failure of this is virtually inconceivable, or because other reactor systems such as light water reactors (LWRs) have special systems, including containment structures, to minimize the release of radioactivity in the event of a failure of the coolant system.

Some of the major events of the last 60 years are described below.

Windscale

One of the earliest reactor accidents occurred at Windscale, Cumbria, UK, in 1957. The reactor was of a very early design and used direct-cycle air cooling, that is, air was drawn in through the reactor core, removing heat, and was discharged back to the atmosphere through filters and a tall stack. A special operation was being performed which caused the fuel rods to overheat and catch fire. The main activity released was iodine-131 which, being a vapour, was not removed very effectively by the filters. An estimated 7×10^{14} Bq of iodine-131 was released and, although evacuation of the local population was not necessary,

milk produced in a large area downwind of the site was declared unfit for consumption. This was because of the exposure pathway:

Iodine-131 → pasture → uptake by cows → milk → consumption of milk → dose to thyroid

Other fission products such as strontium and caesium were also released but, because of the absorptive action of the filters, in much smaller quantities. Retrospectively, the Windscale accident was rated as INES Level 5.

Direct-cycle air cooling is no longer used on power reactors and so further accidents of this type are not possible. A great deal was learned from the Windscale accident about the sort of organization, equipment and procedures that are necessary to deal with major accidents.

Three Mile Island

In 1979, an accident occurred at a large commercial pressurized water reactor (PWR) plant at Three Mile Island (TMI), Pennsylvania, USA. The accident followed a major leak in the pressure system and the failure of a safety system to operate because of an incorrectly aligned valve. The resulting loss of cooling led to the overheating of fuel and release of activity to the environment via the gas waste system. The released activity consisted mainly of short-lived inert fission product gases, so the resulting radiation exposure of the surrounding population was low. However, some evacuation was undertaken, mainly of young children and pregnant women, and the accident had a considerable adverse effect on public attitudes to nuclear power in the USA. As for the Windscale accident, the TMI event was retrospectively given a Level 5 INES rating.

The TMI accident led to increased efforts to understand the processes that would occur during a severe reactor accident. It now seems that, for the majority of accidents that might occur on PWRs, the release of fission products would be considerably less than had previously been thought. This is because various processes within the containment, and the containment structure itself, are very effective in limiting the release.

Chernobyl

A much more serious reactor accident occurred at Chernobyl, Ukraine, in 1986 and involved a 1000-MWe (megawatt electrical) graphite-moderated boiling water reactor, a design peculiar to Eastern Europe. The accident resulted in a major fire and a large release of radioactivity (about 1000 times the amount released in the Windscale accident and a million times the amount released at TMI). Extremely high on-site dose rates occurred and there was extensive contamination, not just in the vicinity of the site but across wide areas of Western Europe. The town of Pripyat (3–5 km from the site), with a population of 45 000, was evacuated in under 3 hours on the afternoon of the second day of the release. At that time, dose rates in the part of Pripyat closest to the site were in the range of 7–10 mSv/h. It has been estimated that the majority of Pripyat's inhabitants received whole-body doses of γ radiation of 15–50 mGy and skin doses of β radiation of 100–200 mGy.

Twenty-five years later, the Chernobyl accident continues to influence worldwide thinking about the radiological consequences of nuclear reactor accidents. Epidemiological studies of the effects of the release on the various affected populations continue to determine treatment regimens, particularly for thyroid cancers in children, and to refine the risk factors. The accident has led to a complete reappraisal of the methods for modelling

the release, transport and uptake of radionuclides, and of the preparations and procedures that are needed to handle such situations. In particular, it has resulted in an International Convention on Nuclear Safety, which is intended to improve the safety of all nuclear plants worldwide, and a Convention on Early Notification of a Nuclear Accident to ensure that all potentially affected countries are notified rapidly of any future nuclear incident. It also provided the main motivation for the INES concept. Retrospectively, it was given the maximum rating of Level 7.

Fukushima

In March 2011, a major earthquake off the east coast of Japan and an associated tsunami led to a serious radiological emergency at the Fukushima nuclear site. Of the six boiling water reactors on the site, three were operating at the time but shut down automatically. The reactor plants and the essential safety equipment generally withstood the effects of the earthquake, as they were designed to do. However, the earthquake caused the loss of an external power connection to the site and the emergency diesel generators started up to supply cooling pumps and other essential equipment. The plants also had sea defence walls to protect against storm conditions and potentially a tsunami, but these proved to be inadequate for the magnitude of the event (the tsunami is estimated to have been about 14 m high when it reached the site). The site was inundated, and this led to loss of the diesel generators and hence of the means of cooling the reactors. The coolant temperatures in the reactors increased, leading to rising pressure and the venting of steam into the primary containment vessel and later into the secondary containment. The fuel became uncovered, leading to melting of the fuel and also to the production of hydrogen from a zirconium steam reaction. Venting of this hydrogen into the containments led to hydrogen explosions which damaged the containments, so providing a path for the escape of fission products into the atmosphere. A further complicating factor arose from loss of cooling to the spent fuel storage ponds, which caused evaporation of water and the uncovering of fuel. The loss of water shielding caused high γ-radiation levels on the site as a result of air scattering and this impeded recovery operations.

Of the several hundred workers struggling to bring the situation under control, a few tens received doses exceeding 100 mSv but lower than the 250 mSv limit set by the authorities for the emergency. The scale of the radioactivity being released to atmosphere caused the authorities to evacuate some 200 000 people living within 20 km of the site and to issue stable iodine tablets over a wider area. It was also necessary to impose a ban on the consumption of locally grown foodstuffs in various areas. Large volumes of contaminated water arose from the efforts to cool the reactor cores and spent fuel ponds, much of which had to be discharged into the sea, resulting in some contamination of marine foodstuffs. Although the radiological impact has not yet been fully evaluated, it is clear that it is very much lower than that from Chernobyl. The remarkable fact is that there was no evidence of any short-term radiation effects in either the on-site workers or the general population. Overall, the health effects of the on-site radiation and the off-site release were very low compared with the wider effects of the disastrous earthquake and the tsunami.

The event was initially classed as INES Level 5 by the Japanese authorities, but this was revised some weeks after the event to Level 7, mainly because all six reactors were affected to some degree. The Fukushima event is unique in that it was the first major nuclear accident to have been caused by an external event (the earthquake) and it is clear that many

important lessons will be learned in the coming years in relation to the protection of plant and equipment essential to reactor safety.

Other potential sources

Other potential sources of a major release of radioactivity are nuclear fuel-reprocessing plants and the waste storage facilities associated with them. As explained in section 11.8.2, after the fuel has been chemically processed, the highly active waste stream, which contains almost all the fission products and higher actinides, is routed into special storage tanks. These tanks may contain several cores' worth of activity and have to be cooled to remove the radioactive decay heat and prevent the build-up of potentially explosive hydrogen gas. Any sustained loss of cooling over many hours, or a severe external event such as an earthquake or the impact of an aircraft, might cause a storage tank to fail and release a significant fraction of its inventory. Such potential events have to be covered in the emergency plans for nuclear fuel-reprocessing plants.

16.5 UNCONTROLLED CRITICALITY

16.5.1 General

The process of fission and the conditions under which a chain reaction can occur have been described in section 11.2. Uncontrolled critical excursions are possible in reactors and in any plant or laboratory in which sufficiently large quantities of fissile materials are handled. The main feature of uncontrolled criticality is the intense prompt neutron and γ radiation given off during the excursion. If it occurs in an area where there is little or no shielding, a very large external hazard results. However, if it occurs in the core of a reactor, the hazard is greatly reduced by the biological shield. In either situation, if the energy released is large enough, it can result in an explosive reaction, loss of containment and a release of radioactivity.

There are three approaches to the prevention of criticality when large quantities of fissile material are present:

1. the provision of **neutron absorbers**;
2. the use of **safe geometry**; and
3. the limitation of quantity (**batching**).

In a reactor, method 1 is the most important while in fuel plants methods 2 and 3 are used either separately or in combination.

16.5.2 Reactors

In a reactor, criticality is maintained by adjusting the position of the control rods (see section 11.3.2). Uncontrolled criticality could potentially occur if the rods failed to enter the core when required or were suddenly ejected from it. The majority of uncontrolled critical excursions that have occurred on reactors have involved low-power experimental facilities rather than large-power reactors. The accidents were, in most cases, caused by a combination of circumstances such as a bad design feature, a mechanical or electrical failure, and an operator error.

The most comprehensively investigated and reported accident was on the SL1 reactor at Idaho Falls, USA, in 1961. Following a routine shut-down for maintenance, an operating

crew of three men was reassembling the control-rod drives in preparation for start-up. The design of the rod-drive mechanisms was such that the rods had to be raised manually a few centimetres while they were being connected. It appears that the central control rod was manually withdrawn about 0.5 m, causing the reactor to go critical. The energy released caused a violent steam explosion which killed the three operators. Recovery operations were hampered by radiation levels of about 10 Sv/h inside the reactor building owing to fission products released from the core. Very little release of radioactivity from the building occurred even though it had not been designed as a containment. The accident resulted from a serious design fault and inadequate supervision or training of the operators. Modern reactor designs, both experimental and power, attempt to ensure that such events are virtually impossible.

16.5.3 Reactor fuel plants

There are three types of reactor fuel plant, namely enrichment, fuel fabrication and irradiated-fuel reprocessing plants, all of which handle large quantities of fissile material. The material may be in solid form or in solution, the latter form being more hazardous because of the neutron moderation provided by the solvent.

The safe geometry method involves making all process vessels, tanks and pipework of such a shape that their contents cannot go critical. The most efficient shape to produce a critical arrangement is a sphere, since in this configuration neutrons are least likely to escape without causing further fission. Conversely, the safest shapes are thin slabs or tall cylinders.

The **safe geometry method** can also be applied to the handling of fissile materials in solid forms such as billets, rods or plates of fuel. An example of this is the **thin-layer method**. Here, the essential feature is that within a given area, all fissile material is stored, processed, transported and generally handled within a certain layer. For example, if the safe thickness for the type of material being handled is 0.15 m, all fissile material would be stored, processed, etc., at a specified height, say between 1.0 m and 1.15 m above floor level. All working surfaces would be 1.0 m high, and trolleys, machines and storage racks would be arranged so that the material always remained in the thin layer.

Batching means that the fissile material is processed through the plant in quantities that are too small to go critical even under the worst geometry. To provide a good margin of safety, batches are usually small enough to ensure safety even if double-batching should occur because of a mechanical or administrative failure. Another important point is that vessels and batches must be adequately spaced to prevent interaction between them.

Whichever method of criticality control is used, an allowance must be made for contingencies. In particular, the possibility of flooding must be considered because of the moderation and reflection provided by water. Plans for fire-fighting are often complicated by the need to preclude water from the area.

Fuel plant accidents are typified by that which occurred at Los Alamos, New Mexico, USA, in 1958. While an inventory was being taken of plutonium residues, the contents of two tanks were drained into a third tank. The two tanks had each contained a safe quantity, but when added together they constituted an unsafe quantity. The residues were fairly heavy and settled in the bottom of the tank in a subcritical configuration. However, when the tank was electrically stirred, the residues mixed with the solvent, which provided neutron moderation, and the system went critical. The operator received a fatal dose estimated at 120 Gy.

Another criticality accident occurred at the fuel reprocessing plant at Tokaimura, Japan, in 1999. The accident was initiated by three inexperienced and inadequately trained operators who added a bucket of enriched uranyl nitrate solution to a process vessel which was already close to criticality. The material in the tank went critical and, although there was no explosion, very high levels of neutron and γ radiation resulted. The system continued to experience intermittent criticality for some 20 hours before being brought under control. The three workers received doses of up to 20 Sv and two of them died, one after 12 weeks and one after 7 months. Other workers on the site, and a small number of members of the public, received doses of up to about 20 mSv. The accident was assessed by the Japanese authorities as Level 4 on INES. It was later concluded that the accident was caused by human error and fundamental breaches of safety principles.

16.6 PRE-PLANNING FOR EMERGENCIES

Whatever the scale of use of radioactive materials or equipment that generates radiation, there is a requirement to undertake a prior risk assessment to identify situations that could give rise to a radiological incident. Contingency plans appropriate to the scale of the potential incidents need to be drawn up. These plans should be included as a component of staff training and should be exercised periodically.

In the case of nuclear facilities, an important result of the TMI accident mentioned in section 16.3.3 was that all countries reviewed their pre-planned arrangements for nuclear emergencies. In the USA, this resulted, among other things, in an increase in the distance from the plant for which detailed pre-planning was required. In the UK, the review group concluded that the distances did not need to be increased but called for a number of actions to improve the coordination of emergency services and to streamline the decision-making process in the event of an emergency. The Chernobyl accident caused a further re-evaluation of emergency arrangements, and the Fukushima event raises wider issues which have yet to be fully evaluated.

Ever since the Chernobyl accident, international organizations such as the EU and national governments and regulatory authorities have adopted an increasingly formal approach to emergency planning and providing information to the public. Increasingly, operators of premises where work with radiation is carried out have been required to carry out formal hazard and risk assessments in order to:

- identify what reasonably practicable measures might be taken to prevent foreseeable accidents and mitigate the consequences of any that might occur;
- provide a structured basis on which to plan for emergencies and provide information to the public;
- supply information to the local authority to enable an off-site plan to be prepared.

In the UK, the Radiation (Emergency Preparedness and Public Information) Regulations (REPPIR) were adopted in 2001 to implement the articles of the 1996 Basic Safety Standards Directive which dealt with intervention in cases of radiological emergencies. REPPIR also implements relevant parts of the 1989 Public Information Directive on informing the general public about health-protection measures to be applied and steps to be taken in the event of an emergency.

As discussed in section 6.5, in emergency situations, doses well above the normal dose limits may be authorized for volunteer workers for the purpose of saving lives or preventing an escalation of the incident. Similarly, for members of the public in the vicinity of the site, decisions on the implementation of countermeasures such as sheltering, evacuation or administration of stable iodine, need to balance the detriments against the expected dose saving. Internationally agreed **intervention levels** (sometimes called **emergency reference levels**) provide broad guidance on the expected dose saving that would be justified by the implementation of countermeasures. In the UK, these are expressed as ranges of values of dose saving so as to allow local circumstances to be taken into account.

Countermeasure	Lower-level mSv	Upper-level mSv
Sheltering	3	30
Evacuation	30	300
Iodine tablets	10	100

In practice, decisions on countermeasures, especially for the population close to the site, may need to be made before adequate information is available on the scale of the release and its potential consequences or even before a release has occurred. Thus early decisions on countermeasures could be based on information about the state of the plant and the likely progression of the fault condition. Pre-planning to deal with emergency situations begins at the design stage of any plant, process or experiment. A detailed safety analysis at this stage not only indicates the major hazards but may also enable methods of reducing them to be incorporated in the design. No matter how good the design or how many safeguards are provided, there always remains the possibility of an accident; thus, an **emergency organization** is required to deal with this possible eventuality.

The size of the organization depends very much on the type of plant and the possible scale of any emergency. In a large plant such as a power reactor, the organization is quite large and includes representatives from different departments. For example:

- **Administration department** can assist with such matters as transport, liaison with external authorities and other services, and communicating with the media.
- **Engineering department** is responsible for providing rescue and damage control teams, decontamination services and maintaining emergency equipment.
- **Medical department** deals with casualties, radiation or otherwise, and liaises with hospitals and medical authorities.
- **Health physics department** provides monitoring equipment and services and advises on all aspects of radiation protection.

The actions required and the responsibilities of the various parties in the organization are detailed in the **emergency procedures**. This document includes instructions on evacuation, monitoring, communications, re-entry and use of emergency equipment.

The emergency equipment includes rescue equipment, medical equipment, protective clothing, breathing apparatus and monitoring instruments. In the last case, it must be borne in mind that very high radiation and contamination levels may occur and so special high-range instruments are required. In the SL1 accident mentioned earlier, the radiation

levels encountered by the rescue team were greater than the maximum range of their instruments (5 Gy/h). Instruments extending up to about 50 Gy/h are now available for emergency use.

Finally, the importance of exercising the emergency organization must be emphasized. No matter what the scale of a potential situation, regular exercises remind staff of their actions and responsibilities, test the emergency equipment and highlight shortcomings in the procedures.

SUMMARY OF KEY POINTS

Radiological incident: abnormal or unexpected radiation hazard.

Various levels of severity: can be localized, have wider on-site effects or have off-site effects.

Radiological emergency: would be declared if an incident requires the implementation of an emergency plan.

INES: internationally agreed scale to denote severity of an event.

Potential causes of radiological emergencies: loss of shielding, loss of containment or criticality – usually as a result of conventional failure.

Detection of situation: vital that an incident is recognized immediately – importance of installed instruments.

Pre-planning begins at design stage and requires detailed safety analysis.

Emergency organization: emergency procedures and equipment. Emergency exercises.

REVISION QUESTIONS

1. What is a radiological incident and how could such a situation arise?
2. When would an incident become an emergency and what would then be initiated?
3. What is the purpose of the International Nuclear Events Scale (INES)? Give examples of INES ratings for past events.
4. Discuss the importance of the rapid detection of an abnormal situation and explain how such detection might be achieved in practice.
5. With particular reference to reactor fuel plants, describe the methods by which the criticality risk is controlled.
6. Write a short set of emergency instructions to apply in the event of a spillage in a small laboratory handling about 100 MBq of a low-toxicity nuclide.
7. List some of the lessons learned from major nuclear accidents of the past 60 years.

17 Legislation and regulations related to radiation protection

17.1 INTRODUCTION

The legislation and regulations dealing with radiological protection vary considerably from one country to another and are often intimately linked with other legislation dealing with, for example, the control of nuclear energy or the protection of the environment. It is not possible to give an account of all the relevant national regulations in this book and readers who require further information on the regulatory position in a particular country should consult the bibliography. However, in most countries the starting point for the regulation of ionizing radiations is the recommendations of the **International Commission on Radiological Protection (ICRP)**. The ICRP's recommendations are incorporated into wider international standards and guidance before eventually finding their way into national legislation. In this chapter, the UK legislative position is considered in some detail and a brief account is given of the most important aspects of the regulatory positions in France, Germany, Japan, the USA, Australia and New Zealand.

17.2 RECOMMENDATIONS OF THE ICRP

The fact that the regulation of ionizing radiations around the world is remarkably consistent is due in large measure to the ICRP. The ICRP is a non-governmental scientific organization which has published recommendations about protection against ionizing radiations for over 80 years. Its authority is recognized worldwide and governments evaluate its recommendations and put them into practice in ways appropriate to their circumstances. The most recent recommendations of the ICRP were issued in 2007 in *Publication 103* (as discussed in Chapter 6). These have been incorporated into revised Basic Safety Standards for Protection against Ionizing Radiation and for the Safety of Radiation Sources which was approved by the International Atomic Energy Agency in September 2011. The basic safety standards are co-sponsored by the IAEA, the World Health Organization (WHO), the International Labour Organization (ILO), the Nuclear Energy Agency of the Organization for Economic Cooperation and Development (OECD-NEA), the Food and Agriculture Organization of the United Nations (FAO), the Pan-American Health Organization (PAHO), the United Nations Environment Programme (UNEP) and the European Commission (EC).

At present, most countries have legislation which is based on the 1990 recommendations of the ICRP in *Publication 60* and the earlier version of the basic safety standards – IAEA *Safety Series 115* published in 1996. The UK, as a member of the European Union (EU), follows the procedures required of member states by the relevant treaties.

17.3 THE EURATOM DIRECTIVE

Members of the EU are subject to the provisions of the treaty establishing the **European Atomic Energy Community (Euratom)**. A requirement of this treaty is that basic standards should be laid down in the community for the protection of workers and the public against the dangers arising from ionizing radiations. These standards are promulgated via a Basic Safety Standards Directive, the contents of which must be given effect in the member states. The most recent directive, adopted in 1996, takes account of the recommendations in ICRP *Publication 60*. The Directive is currently being revised following publication of ICRP *103*.

17.4 CONVERTING THE EC DIRECTIVE INTO UK LAW

In the UK, most of the requirements of the Basic Safety Standards Directive (BSSD) are fulfilled by the Ionizing Radiations Regulations (1999), which came into effect on 1 January 2000 and which were drafted under the **Health and Safety at Work (HSW) Act (1974)**. The requirements that are not fulfilled in this way are generally implemented within other legislation, such as the Environmental Permitting Regulations 2010 (see section 17.6) and the Ionizing Radiation (Medical Exposure) Regulations 2000 (see section 14.2). The HSW Act is a major piece of legislation which was enacted with the aim of rationalizing the existing, rather fragmentary, legislation on the health and safety of persons at work and of other persons who may be put at some risk by the activities of persons at work. The Act provided for the setting up of a **Health and Safety Commission (HSC)**, which consists of representatives of employers' organizations, trade unions and other organizations, such as local authorities. The HSC is responsible under the Act for formulating regulations, disseminating information to employers, employees and others, making arrangements for carrying out any necessary research and training, and generally providing advice on the interpretation of the Act. The Commission reports to the particular Secretary of State who is responsible for the activity or work area in question. The actual administration of the statutory requirements of the Act is the responsibility of the **Health and Safety Executive (HSE)**, which comprises three persons appointed by the Commission. It is the duty of the HSE to make adequate arrangements for the enforcement of the relevant statutory provisions, except where such enforcement is transferred to some other authority (e.g. local authorities). The HSE enforces the various regulations through a number of organizations which have been incorporated into the overall organization, for example the Factories Inspectorate, the Office for Nuclear Regulation, etc. The HSW Act is a very extensive and complicated piece of legislation and persons with responsibilities subject to the Act are advised to read the detailed regulations.

17.5 REGULATORY FRAMEWORK UNDER THE HEALTH AND SAFETY AT WORK ACT

The regulatory framework for radiological protection adopted by the HSC in 1999 consisted of a package which comprised the **Ionizing Radiations Regulations (1999)** and an approved code of practice entitled *Work with Ionizing Radiation*. This approved code, with its associated guidance, gives detailed advice on the scope and duties of the requirements imposed by the Ionizing Radiations Regulations (1999).

17.5.1 Regulations

The basic principle in the 1999 Regulations is that all necessary steps should be taken to reduce, so far as is reasonably practicable, the extent to which people are exposed to ionizing radiation. This means that it is not sufficient merely to observe the dose limits specified in the regulations; there is a further requirement to weigh the costs of the possible health detriment from exposure against the costs of reducing or eliminating that exposure. In drawing up plans to restrict exposure, the regulations require the use of dose constraints where appropriate. **Dose constraints**, as a tool to assist the process of optimization, were introduced in ICRP *Publication 60*. They are intended to be used solely in a prospective way to help ensure, at the design stage, that no new practice 'uses up' a disproportionate amount of the relevant dose limit. Although useful in theory, experience has shown that it is difficult to avoid confusing constraints with dose limits and investigation levels, and ICRP *Publication 103* has sought to provide additional clarification.

The **dose limits** specified in the regulations for various categories of persons (workers, trainees, members of the public, women of reproductive capacity) refer to the sum of all radiation absorbed and committed from both external and internal sources, whether to the whole body or part of the body, arising from work activities. The basic dose limit specified in the regulations for the effective dose to any employee aged 18 years or above is 20 mSv in any calendar year. However, where an employer is able to demonstrate that this dose limit is impracticable, an effective dose limit of 100 mSv in any period of five consecutive years may be used, subject to a maximum effective dose of 50 mSv in any single calendar year. In practice, UK employers have generally adopted the 20 mSv annual effective dose limit, with a lower limit (6 mSv) for trainees under 18 years of age. The limit on effective dose for any other person is set at 1 mSv in any calendar year.

Individual dose limits for the lens of the eye, skin and hands, forearms, feet and ankles are also specified. The limit on equivalent dose for the abdomen of a woman of reproductive capacity who is at work is 13 mSv in any consecutive period of 3 months.

As regards medical exposures, the doses received by patients undergoing diagnosis or treatment involving the use of ionizing radiation are not taken into account in determining compliance with dose limits. However, doses received by patients as a result of other patients' medical exposures are taken into account.

In addition, the regulations specify an **investigation level of dose**. This is nominally set at three-quarters of the annual dose limits for workers aged 18 years or over. However, the radiation employer in certain sectors may decide to set this dose investigation level at a lower value depending on the work carried out (e.g. in hospitals, it can be typically one-tenth of the annual dose limits). If this dose investigation level is exceeded during the course of any calendar year, an investigation must be carried out to ensure that doses are being kept as low as reasonably practicable and to initiate remedial measures if they are not.

The regulations place requirements with respect to the health surveillance, dose assessment and record-keeping requirements for workers who have been designated as classified persons by their employer. **Classified persons** are designated as such because they are likely to receive an occupational effective dose of 6 mSv or more in any calendar year.

To facilitate the **control of doses**, the regulations require controlled and supervised areas to be identified where persons need to follow special procedures to restrict exposure or where there is a likelihood of receiving equivalent doses in excess of three-tenths and

one-tenth respectively of the annual dose limit for workers aged 18 years or over. Any person entering a controlled area must be designated a classified person unless he or she enters under a written system of work designed to ensure that no significant dose can be received.

There are requirements for people to notify the HSE when they use ionizing radiation and to carry out a prior assessment of the risks arising from their work with ionizing radiations (see section 17.5.3). The HSE must also be notified where there is a significant unplanned release, loss or theft of radioactive substances and when someone has received an excessive dose of radiation. Local investigations of excessive doses have to be made and **records** kept.

The provision of information on potential hazards and the instruction and training of people involved with ionizing radiation are required by the regulations. In addition, there are requirements to formulate written **local rules** covering all radiation protection arrangements and to provide supervision of work involving ionizing radiation; this normally necessitates the appointment of a **radiation protection supervisor** (**RPS**). Furthermore, the regulations require **radiation protection advisers** (**RPAs**) to be appointed whenever expert advice is needed and specifically whenever an employer has to designate one or more controlled areas.

Finally there are requirements to:

- provide appropriate safety devices, warning signals, handling tools, etc.;
- leak test radioactive sources;
- provide protective equipment and clothing and test them;
- monitor radiation and contamination levels;
- store radioactive substances safely;
- design, construct and maintain buildings, fittings and equipment so as to minimize contamination;
- make contingency arrangements for dealing with foreseeable but unintended incidents.

17.5.2 Approved code of practice and other guidance

The Ionizing Radiations Regulations (1999) contain the fundamental requirements for control of exposure to ionizing radiation. Details of acceptable methods of meeting those requirements are given in the supporting **Approved Code of Practice**. An approved code of practice does not, in itself, have any legal standing, but it would be taken into account by a court when deciding whether or not a person or organization had failed to comply with the regulations under the HSW Act. The code of practice is couched in much more understandable terms than the regulations governing it and it can explain the reason for the procedure which is included and thus aid the reader's understanding.

17.5.3 Prior risk assessment

Regulation 7 of the Ionizing Radiations Regulations (1999) requires that:

> *before a radiation employer commences a new activity involving work with ionizing radiation in respect of which no risk assessment has been made by him, he shall make a suitable and sufficient assessment of the risk to any employee and other person for the purpose of identifying the measures he needs to take to restrict the exposure of that employee or other person to ionizing radiation.*

This means that, before any new radiation work commences, the responsible person, usually the work supervisor, must ensure that a risk assessment is made which identifies the hazards and evaluates the risks to both workers and any other persons who may be exposed. This risk assessment must be 'suitable and sufficient' and, to ensure this, the responsible person should adopt a systematic approach, which is almost invariably helped by some degree of quantification. However, it is important to realize that quantification cannot remove the inherent uncertainty associated with any risk and this needs to be recognized when formulating risk management strategies.

The process of risk assessment has been discussed in more detail in Chapter 15.

17.6 ENVIRONMENTAL REGULATORY FRAMEWORK IN THE UK

The requirements of the BSSD that relate to the control of radioactive material and any subsequent disposal of radioactive waste were originally implemented in the UK within the **Radioactive Substances Act 1993 (RSA93)**. RSA93 has subsequently been repealed in England and Wales following its incorporation into the **Environmental Permitting Regulations 2010 (EPR2010)**, although the legislative requirements effectively remained unchanged.

EPR2010 is enforced by the Environment Agency (EA) and the using and keeping of radioactive material and the accumulation and disposal of radioactive waste is regulated by the issuing of permits with associated conditions. Similarly, the Scottish Environment Protection Agency (SEPA) and the Northern Ireland Environment Agency (NIEA) enforce RSA93 and issue registrations for the using and keeping of radioactive material and authorisations for the accumulation and disposal of radioactive waste in Scotland and Northern Ireland, respectively.

Radioactive Substances Exemption Orders 2011 for Scotland and Northern Ireland and the **Environmental Permitting (England and Wales) Amendment Regulations 2011** make provision to exempt certain radioactive materials from the requirements of RSA93 and EPR2010. Such provision has been made within the regulatory framework as it is deemed that there is no net benefit in regulating exempt materials owing to the very low radiation risk that they present.

17.7 TRANSPORT OF RADIOACTIVE MATERIAL

The International Atomic Energy Authority (IAEA) has recommended that all member states use its **Regulations for the Safe Transport of Radioactive Material (TS-R-1)** as a basis for corresponding national and international legislation to ensure that radioactive material transported across different countries is regulated consistently along its route. As such, the regulations are generally written in a prescriptive manner; for example, they specify the package testing and labelling requirements for different types of transport packages.

The IAEA Regulations have been adopted in Europe via a European agreement concerning the International Carriage of Dangerous Goods by Road (ADR) and Regulations Concerning the International Transport of Dangerous Goods by Rail (RID). These have been implemented in the UK by the Carriage of Dangerous Goods and Use of Transportable Pressure Receptacles Regulations 2009 (CDG Regs), which generally reference back to the ADR/RID.

17.8 OTHER RELEVANT UK LEGISLATION

There are several other pieces of UK legislation which have an impact on radiation protection (see Table 17.1).

Table 17.1 Summary of UK regulations

Regulation	Main provisions
Atomic Energy Act (1946)	Promotion and control of atomic energy development
Nuclear Installations Act (1965, 1969)	Deals with licensing and insurance of specified sites which have substantial radiological hazards
Health Protection Agency Act (2004)	The National Radiological Protection Board (NRPB) and Advisory Committee set up following the enactment of the Radiological Protection Act became the Radiation Protection Division of the Health Protection Agency (HPA)
Health and Safety at Work, etc. Act (1974)	Set up Health and Safety Commission and Health and Safety Executive to administer regulations concerning the health and safety of persons at work, and of other persons who may be put at risk by the activities of persons at work
The Ionizing Radiations Regulations 1999	The regulatory package consists of the regulations and an approved code of practice plus more detailed guidance
The Ionizing Radiation (Medical Exposure) Regulations 2000	Apply to the radiation exposure of individual patients undergoing medical diagnosis or treatment. Also cover radiation exposures as part of occupational and health screening programmes, medico-legal exposures and research projects which expose volunteers to ionizing radiation
Radiation Emergency Preparedness and Public Information Regulations 2001	Apply to organizations that hold significant quantities of unsealed radioactive materials. Provide protection to members of the public from emergencies that might arise from work with ionizing radiations
The Radioactive Substances Act 1993	Applies to the keeping of radioactive material and the accumulation and disposal of radioactive waste (Scotland and Northern Ireland)
The Radioactive Substances Exemption Order (Scotland) 2011; The Radioactive Substances Exemption Order (Northern Ireland) 2011	Exemptions relating to the requirements of the Radioactive Substances Act
The Environmental Permitting (England and Wales) Regulations 2010	Regulations on keeping and use of radioactive material and the disposal of radioactive waste. The Radioactive Substances Act 1993 was migrated into Schedule 23 of the Environmental Permitting Regulations
The Environmental Permitting (England and Wales) Amendment Regulations 2011	Exemptions relating to the requirements of the Environmental Permitting (England and Wales) Regulations in relation to the control and disposal of radioactive material
Carriage of Dangerous Goods and use of Transportable Pressure Equipment Regulations 2009	Regulations on the road transport of radioactive material, based on the International Atomic Energy Agency's (IAEA) Regulations for the Safe Transport of Radioactive Material (TS-R-1). Specific details of the requirements for transport are not given in the regulations; instead the operator must refer to the European Agreement concerning the International Carriage of Dangerous Goods by Road (ADR) 2011

17.9 BRIEF SUMMARY OF INTERNATIONAL GUIDANCE AND REGULATIONS IN OTHER COUNTRIES

In section 17.3, the EU directive on radiological protection was discussed. This directive lays down the basic standards for the protection of workers and the public against the dangers arising from ionizing radiations within member states. It is binding on member states with regard to the results which have to be achieved. The Euratom treaty also requires member states to make appropriate provisions to ensure compliance with the basic standards and it also requires that provisions on radiological protection be communicated to the commission to achieve the harmonization of such provisions in member states.

Other organizations which undertake work at international level to formulate radiation protection standards are:

- The **International Commission on Radiological Protection** (ICRP) (see Chapter 6).
- The **International Atomic Energy Agency**, which produces Basic Standards for Radiation Protection, in collaboration with other international organizations (the governments of member states are invited to use these as a basis for formulating national legislation on radiation protection). In addition, IAEA has published a number of expert studies in the form of codes of practice, which give practical safety guidelines for specific applications.
- The **World Health Organization (WHO)** has published various reports concerning radiological protection and has collaborated with IAEA and other organizations in producing a number of guides on the protection of the public and workers against nuclear hazards. However, the WHO has not issued any regulations in this field.
- In 1960, the **International Labour Organization (ILO)** adopted a convention concerning the protection of workers against ionizing radiations. In 1963, the ILO published a manual of industrial radiation protection which contains a model code of safety regulations concerning ionizing radiations, plus several guides for specific areas.
- The **Nuclear Energy Agency** (NEA) of the OECD issues radiation protection norms from time to time. These norms are based on ICRP recommendations and are revised periodically to take account of the latest ICRP recommendations. In addition, NEA has sponsored or participated in the preparation of a number of studies in the field of radiological safety and collaborates with other international organizations in the establishment of safety standards.

The legislation and regulations dealing with radiological protection in individual countries are quite diverse and complicated, and it is beyond the scope of this book to attempt a discussion of them. Readers who wish to pursue this topic further should consult the bibliography. However, the regulatory frameworks in France, Germany, Japan, the USA, Australia and New Zealand are briefly outlined below.

17.9.1 France

In general, the legislative framework relating to radiological protection has evolved in line with technological advances in the use of nuclear power in France, and many of the legal

provisions can be found within general legislation. Radiological standards are based on the EU BSSD, which in turn derives from the recommendations of the ICRP.

The main radiation protection legislation can be found in:

Protection of workers

- Labour Code, Legislative Part and Regulatory Part (Ionizing radiation)
- Decree No. 75-306, 28 April 1975, which relates to the protection of workers in nuclear installations
- Decree No. 86-1103, 2 October 1986, which relates to the protection of workers from ionizing radiation hazards

Protection of the public

- Public Health Code, Legislative Part and Regulatory Part (Ionizing radiation).

The DGSNR (Direction Général de la Sûreté Nucléaire et de la Radioprotection) reporting to the Ministers for Industry, Health and Ecology and Sustainable Development is the lead directorate which has responsibilities for nuclear safety and radiation protection. It is assisted by the IRSN (Institut de Radio-Protection et de Sûreté Nucléaire).

In addition, the longstanding Commissariat à l'Energie Atomique (CEA), originally set up by the government in 1945, has the main task of developing the use of radioactive material for the purposes of energy, health, defence and industry.

17.9.2 Germany

The legal basis for radiation protection in Germany is the Atomic Energy Act 1959. This governs the Radiation Protection Ordinance 2001 and the X-ray Ordinance 2002, which enable the implementation of the EU BSSD.

The lead government department for radiation protection, the **Federal Ministry for the Environment, Nature Conservation and Nuclear Safety (BMU – Bundesumweltministerium)**, was founded in 1986, a matter of weeks after the Chernobyl nuclear accident. This was aimed at bringing the responsibility for radiation emergency preparedness under a single ministry. At the same time, the Precautionary Radiation Protection Act 1986 (amended in 2008) was enacted with the purpose of protecting the public by monitoring the environment in the event of a radiological emergency.

The **Federal Office for Radiation Protection (BfS – Bundesamt für Strahlenschutz)** is a scientific and technical Superior Federal Authority which functions under the umbrella of the BMU. It was founded in 1989 and is mainly occupied with the implementation and enforcement tasks in accordance with the Atomic Energy Act and the Precautionary Radiation Protection Act.

The BMU is supported by an independent advisory body, the **Commission on Radiological Protection (SSK)**, which provides advice and recommendations with respect to the radiation safety of workers and members of the public.

17.9.3 Japan

The Atomic Energy Act of 1955 (latest revision 2004) lays down the main principles governing protection against radiation hazards. Under this Act there are other laws such as The Law Concerning Prevention of Radiation Hazards due to Radioisotopes etc. (1955), which is aimed mainly at ensuring public safety by preventing radiation hazards

that may arise from radioisotopes and apparatus generating ionizing radiation, and The Special Law for Nuclear Disaster Measures (1999), which relates to nuclear emergency preparedness including the requirement for a coordinated response at both the local and national level.

In Japan, the control of nuclear energy is quite centralized. The **Radiation Council** is responsible for drafting and updating radiation protection legislation which is based on ICRP recommendations. The **Nuclear Safety Commission (NSC)**, which is supported by the Special Committee on Radiation Protection, is appointed directly by the Prime Minister to act as the administrative body responsible for supervising and auditing the activities of the **Nuclear and Industrial Safety Agency (NISA)**. NISA is the Japanese regulatory body and reports directly to the Minister for Economy, Trade and Industry (METI).

17.9.4 United States of America

In the USA, the **Nuclear Regulatory Commission (NRC)** has the prime responsibility for matters concerned with the civil nuclear reactor programmes. The detailed requirements are set out in the Code of Federal Regulations, Title 10, *Energy*, known as 10CFR. Part 20 of 10CFR, Standard for Protection Against Radiation, sets out permissible doses and levels of radioactivity in effluents. Under the Atomic Energy Act, the NRC may devolve some of its regulatory responsibilities to a state through written agreements. 'Agreement States' must then operate programmes that are the same or more restrictive than the NRC's regulations. However, under these agreements it is not possible for the NRC to devolve the regulation of nuclear power stations.

Like the NRC, the **Department of Energy (DOE)** originated from the Atomic Energy Commission. It is responsible for energy research and the military nuclear programmes. The Office of Worker Safety and Health Policy (part of the DOE) is responsible for developing and issuing the DOE's occupational radiation protection policy requirements, which are set out in Part 835 of 10CFR.

Under the provisions of the National Environmental Policy Act of 1969, the **Environmental Protection Agency** has responsibilities in relation to the environmental impact of nuclear energy, which is regulated by a number of acts such as the Nuclear Waste Policy Act, the Low Level Radioactive Waste Policy Act and the Indoor Radon Abatement Act.

17.9.5 Australia

The Australian Radiation Protection and Nuclear Safety (ARPANS) Act 1998 established the **Australian Radiation Protection and Nuclear Safety Agency (ARPANSA)** as the Federal Government body with the responsibility to regulate the use of radioactive material by Australian Commonwealth Government entities. This specifically includes the regulation of Australia's only research reactor and the development of nuclear technologies by the Australian Nuclear Science and Technology Organisation (ANSTO). ARPANSA has produced various codes and standards based on the IAEA Basic Safety Standards 1996 that form part of its regulatory framework.

Individual states and territories are responsible for the regulation of the use of radioactive material by non-Commonwealth Government entities. Historically, this approach has led to each state being governed by slightly differing sets of legislation, for example the Radiation Safety Act 1999 (Queensland) and Radiation Control Act (New

South Wales). This issue was addressed in 2004 when the first edition of the *National Directory for Radiation Protection* was endorsed at ministerial level. This provides a uniform national framework for radiation protection and includes a commitment by states to adopt ARPANSA's codes and standards.

The ARPANS Act established several bodies to provide advice to ARPANSA. These include the Radiation Health Committee, the Nuclear Safety Committee, and the Radiation Health and Advisory Council.

17.9.6 New Zealand

The primary legislation relating to the use of radioactive material in New Zealand is the Radiation Protection Act 1965 and the Radiation Protection Regulations 1982. This legislation requires users of radioactive material or radiation generators to be issued with a licence in relation to their particular use of ionizing radiation. As part of the licence conditions, licensees are required to meet the requirements of Codes of Safe Practice, for example NRLC7 is the Code of Safe Practice for the Use of X-rays in Dentistry.

The **National Radiation Laboratory (NRL)**, which is a unit of the Ministry of Health, is responsible for regulation with respect to radiation protection. In addition to its regulatory function, NRL maintains an emergency response capability and provides services such as the calibration of radiation protection monitoring instrumentation, provision of a dosimetry service, radiation protection advice and monitoring of radioactivity in the environment.

The Director of NRL is a member of the **Radiation Protection Advisory Council**, which is responsible for providing advice and making recommendations to the Minister of Health or the Director-General of Health on Radiation Protection Matters.

SUMMARY OF KEY POINTS

ICRP: non-governmental scientific organization that publishes recommendations on protection against ionizing radiations. Most recent recommendations in ICRP *Publication 103* (2007).

Basic Safety Standards Directive: sets standards for radiation protection of workers and members of public which must be implemented in all EU member states. Most recent Directive adopted in 1996.

Ionizing Radiations Regulations (1999) converted most of the requirements of the Basic Safety Standards Directive into UK law. Supplemented by an Approved Code of Practice and other guidance.

Environmental Permitting Regulations (2010) converted the requirements of the Basic Safety Standards Directive with respect to the control of radioactive material and to the accumulation and disposal of radioactive waste into law in England and Wales. The Radioactive Substances Act 1993 is the equivalent legislation in Scotland and Northern Ireland.

Prior risk assessment: requirement of Ionizing Radiations Regulations (1999) that risk assessment must be carried out before any new radiation work commences. Must be suitable and sufficient.

REVISION QUESTIONS

1. Discuss the role of the ICRP in the regulation of ionizing radiations and explain why further steps are needed to incorporate ICRP's recommendations into national regulatory frameworks.
2. Discuss the relationship between the Basic Safety Standards Directive (1996) and the Ionizing Radiations Regulations (1999).
3. Explain what is meant by the term 'dose constraint'.
4. What are the dose limits specified in the Ionizing Radiations Regulations (1999) for workers and members of the public?
5. Discuss the relationship between the Basic Safety Standards Directive (1996) and the Environmental Permitting (England and Wales) Regulations 2010.
6. Discuss the reason for all IAEA member states using IAEA regulations as the basis of their national legislation for the transport of radioactive material.

18 The organization and administration of radiation protection services

18.1 THE OVERALL PROCESS

The previous chapters of the book have been devoted to explaining the technical concepts and practice of radiation protection. In this final chapter, the more general aspects of organization and administrative control are considered.

The process of radiation protection depends on:

1. the establishment of standards for radiation protection;
2. the formulation of regulations and codes of practice to meet the standards;
3. the design and operation of plants or facilities in accordance with the codes;
4. the continuous review and audit of the whole process.

18.2 STANDARDS, REGULATIONS AND CODES

It has been mentioned in earlier chapters that the radiation protection standards used in most countries are based on the recommendations of the International Commission on Radiological Protection (ICRP). These recommendations are continually reviewed in the light of new evidence from research institutions in many parts of the world. As explained in Chapter 17, countries that are members of the European Union (EU) are subject to the Euratom Treaty and must comply with the standards laid down in the EU Directives on radiation protection. The Directives reflect the thinking and numerical values of the most recent ICRP recommendations. At the time of writing, the relevant Directive is Council Directive 96/29/EURATOM of 13 May 1996, but a revised Directive reflecting the 2007 recommendations of the ICRP is in preparation. However, it will take some years for national legislation to be revised in the light of the new Directive.

In the UK, the 1996 Directive is implemented by a comprehensive regulatory framework established under the Health and Safety at Work (HSW) Act. A multi-tiered approach was adopted which comprised:

1. **Regulations,** particularly the Ionizing Radiations Regulations (1999), which specify duties and requirements, prior risk assessments, dose limits, classification of persons and areas, record-keeping requirements, medical surveillance and so on.
2. An **Approved Code of Practice**, which explains the detailed application of the regulations in specific areas of work activity and gives guidance on matters

of general concern to those involved with ionizing radiations. This code of practice is issued by the Health and Safety Executive (HSE).

3. **Guidance notes**, published under the auspices of the HSE, which give more detailed advice and guidance on certain topics, taking into account local circumstances. At the plant level, there are usually detailed procedure documents, prepared within the user organization, which interpret the requirements of the official regulations in the context of the particular operations being undertaken. There is also a separate set of medical and dental guidance notes prepared by the Institute of Physics and Engineering in Medicine (IPEM).

18.3 DESIGN AND OPERATION

Radiation protection in any facility begins at the planning and design stage. This means that the designer must be familiar with the general concepts of radiation hazards and the means of controlling them. In addition, there must be extensive consultation between the designer and various specialists. These could include chemists, physicists, health physicists, radiation protection advisers, nuclear safety specialists, fire prevention officers and industrial safety experts.

The need for **advance planning** applies not only to the design of the plant but also to the methods of operation and maintenance. A risk assessment should be undertaken and where this shows that an error in operation could result in a serious radiation hazard, engineered safeguards should be provided either to prevent the error or to limit its consequences. For example, for a reactor, the consequences of an operator error are limited by the reactor protection system, which would shut down the reactor if the power or temperature exceeded some pre-determined value.

The problems of **maintenance** must also be given careful consideration at the design stage. The sort of situation that should be avoided is the location of equipment which needs frequent maintenance in areas of high radiation. Similarly, the possibility of releases of radioactive contamination as a result of maintenance operations and the implications for facility layout should be considered. All of these aspects are considered as part of a design **review process**, often called an ALARA (as low as reasonably achievable) review. This involves detailed assessments of the radiological conditions that will arise in the operation of the facility and of the requirements for access to the various areas and items of equipment for the purposes of operation and maintenance. The major contributions to worker dose and ways of reducing this dose can then be identified. The review usually follows a formal approach addressing a series of issues such as 'Can the source terms be reduced by reducing the amount of radioactivity in a part of the system?'; 'Can shielding be provided on part or all of the active plant?'; 'Can the layout be improved to increase the distance?'; 'Can the operating procedures be changed to reduce the amount of access needed?'; 'Can maintenance requirements be reduced by careful selection of equipment?'. Account also needs to be taken of the possibility of abnormal operating or fault conditions and of incidents. The results of such reviews are fed back into the design in an iterative process. In large and complex facilities such as reactors or chemical plants, operation and maintenance is usually in accordance with detailed written procedures. These are subject to periodic review.

18.4 REVIEW AND AUDIT

Review and audit of the processes of radiation protection occur at several levels. At the top level, the ICRP keeps under continuous review the overall philosophy for radiation protection in the light of advancing knowledge and changing circumstances. It also keeps under review information on the biological effects of radiation in order to ensure that its recommendations remain valid.

At the national level, government bodies or regulators maintain a watching brief on the industries for which they have responsibilities. Statistical surveys can be an important component of these reviews and allow the safety performance of operators to be audited. For example, annual reviews of dose statistics broken down into different sectors and occupations can help to reveal trends and identify areas that need special attention. Similarly, event reports and accident statistics can focus attention onto problem areas. At the organizational level, periodic reviews of radiation protection are essential. For example, the preparation of an annual radiological safety report provides the staff responsible for preparing the report with an opportunity to pause and take stock. The report also helps the management to monitor the situation.

At a more detailed level, the health physics staff with responsibilities for a facility need to keep under continuous review a whole range of issues. These include:

- the radiological conditions around the facilities, particularly during non-routine operations and maintenance periods;
- the adequacy of the classification of controlled and supervised areas;
- the dose performance of different groups of workers;
- the selection, calibration and maintenance of monitoring equipment;
- the training of health physics and other staff.

All of these review processes are an important part of the overall system of radiation protection and allow risk assessments to be periodically updated. They ensure that the appropriate safety standards are applied, that regulatory policies and procedures develop with changing circumstances and that good practice is applied to all aspects of the design and operation of facilities.

18.5 THE HEALTH PHYSICS ORGANIZATION

A **health physics department** is strictly a service organization. Its function is to advise on all matters relating to radiation safety and to provide personnel and equipment to ensure that safety standards are being met. Its attitude should be proactive in that its staff should try to anticipate problems and suggest alternative approaches rather than waiting for problems to arise and then 'stopping the job'. This presupposes good communications at all levels with other departments.

In general, members of a health physics organization should be independent and not have other duties in which a conflict of interests could occur. This is not always possible in small organizations where health physics duties may be a part-time responsibility, but the job specifications should be carefully constructed to avoid potential conflicts.

The 1996 Basic Safety Standards Directive (BSSD) defines the term **'qualified expert'** and establishes requirements for the training, experience and recognition of such experts.

An Annex to the Directive gives the topics to be addressed in a basic syllabus for the education in radiation protection of the qualified expert. In addition, it recommends specific topics to be included in the syllabus for five specific areas, that is, nuclear installations, general industry, research and training, medical applications and accelerators. Surveys within the EU indicate a wide diversity in the approaches of Member States to the training and qualifications of the radiation protection expert. This makes mutual recognition of the qualified expert, as defined in Article 1 of the BSSD, difficult between Member States. Steps are being taken to improve this situation, including the creation of a discussion platform to allow for a better harmonization of education and training requirements in the different areas of radiation protection.

In the UK, the term **Radiation Protection Advisor (RPA)**, as defined in the Ionizing Radiations Regulations, is used for a radiation protection professional who has clearly demonstrated to an approving body that they are competent in most of the 'qualified expert' syllabus of training and experience. Employers whose operations require the designation of **controlled** or **supervised areas** (see section 17.5.1) must appoint a suitably experienced RPA who holds a certificate of competence issued by an HSE-approved RPA assessment body (e.g. RPA2000). The other parts of the 'qualified expert' syllabus of training are fulfilled by the requirements for **radioactive waste advisors (RWAs)** as required under the Environmental Permitting Regulations.

In addition to the RPA, a **Radiation Protection Supervisor (RPS)** should be appointed by the radiation employer to provide local supervision. The RPS should be suitably qualified and should have a sound working knowledge of radiological protection in relation to the particular activities for which he or she is appointed. More than one RPA and RPS may be appointed if the range of the duties or the processes involve warrants.

In addition to the RPA and the RPS, the radiation protection organization of a large establishment will usually contain technical support staff, health physics supervisors, health physics surveyors and administrative support staff.

The basis of an efficient radiation protection organization is a sound programme of routine work. This includes:

1. the administration of personnel monitoring services and the keeping of up-to-date dose records;
2. the performance and recording of routine radiation and contamination surveys in and around controlled areas, and the analysis of the results to observe trends;
3. quality control checks of radiation-generating equipment and the associated control devices;
4. the provision and maintenance of calibrated radiation-monitoring equipment;
5. the mustering, leakage testing and accounting of all radioactive sources;
6. provision and regular testing of emergency equipment;
7. provision of radiation protection training.

There may be a number of other duties depending on the range of activities and organization of the establishment as a whole, for example:

8. provision of protective clothing and equipment;
9. control of radioactive waste and the keeping of records; and
10. making arrangements for the medical surveillance of classified persons.

Then there are, of course, the day-to-day operational aspects, including the provision of advice and monitoring during special or non-routine operations in which significant radiological hazards could arise.

18.6 DOCUMENTS AND REPORTS

Various records and reports are required to be kept either because of statutory obligation or in accordance with a code of practice.

The **health record** is a record of all medical examinations performed on an employee while they are a classified person. The records for a particular person are required to be preserved for 50 years after the last entry.

The **source record** contains information on all radioactive sources and the dates and results of all leakage tests. Retention period is 2 years after the last entry.

The **instrument record** is used to record details and results of tests of all instruments used for health physics purposes. The retention period is 2 years after the last entry.

Radiation dose records are required to be kept in respect of all classified persons and to be retained until the person reaches 75 years of age or for 50 years after the last entry, whichever is the greatest. Under the HSE regulatory framework, an Approved Dosimetry Service (ADS) has to be used for the assessment of personal dose and the retention of dose records for classified persons (i.e. workers who are likely to receive doses greater than three-tenths of any dose limit). Each record includes the worker's national insurance number, and the worker has the right to examine his or her dose record. When a worker changes jobs, he or she will be given a termination record detailing the current dose status (purely for information) and the new employer will seek such a record from his or her own ADS.

Radiation passbooks are used to ensure that the doses received by itinerant workers are kept within the statutory dose limits. Employers must have arrangements for them to be issued on an individual basis to outside workers and to ensure that they are kept up to date. When the worker moves to a different employer, he or she transfers the passbook to that employer.

Survey records, containing the results of routine radiation and contamination surveys, are usually retained for at least 2 years.

Reports of unusual occurrences: all unusual occurrences that result in or could have resulted in an abnormal radiation dose (i.e. above the Dose Investigation Level of dose set by the radiation employer) should be fully investigated. The reports are normally preserved for at least 50 years. Similarly, all reports following investigations into accidental releases or spillages should be retained for 50 years.

Waste disposal records are preserved indefinitely, in particular those records showing the location of buried solid waste.

18.7 TRAINING

The safety records of industries that use sources of ionizing radiation, particularly the nuclear industry, are generally very good. This is, to a large extent, the result of their positive attitude towards **staff training**. The legislation and codes of practice require that all persons exposed to radiation in the course of their work should be given training in the hazards and the means of controlling them. This may vary from a short talk on the function

of personal dosimeters and an outline of local rules to a detailed course in radiation protection, depending on the nature of the facility and the duties of those involved.

Specialist courses are necessary for radiation protection personnel. Large establishments often run their own training courses for health physics surveyors, radiation protection supervisors and health physicists, while others make use of courses run by colleges or universities or by specialist organizations. In the UK, the **Radiological Protection Division** of the Health Protection Agency organizes a number of courses on radiological protection, ranging from short familiarization courses to advanced courses for specialist radiation protection professionals and health physicists. In North America and Europe, there are well-established summer schools that aim to provide intensive refresher courses for professionals in radiological protection.

In the nuclear industry, a most important aspect of training for all health physics personnel is plant familiarization, that is, instruction in the processes and engineering aspects of the plant on which they will be working. An understanding of the plant is essential if proper advice is to be given and a constructive attitude maintained.

It should be noted that records of all staff training must also be retained by the radiation employer.

SUMMARY OF KEY POINTS

ALARA review: a formal review of the design and/or operation of a facility.

Considerations in routine radiation protection: standards, regulations and codes, design and operation of the facility, radiation protection organization, documentation and reporting.

Standards are based on the recommendations of the ICRP.

Regulations are set at national level. In the European Union, regulations must comply with the Basic Safety Standards Directive which reflects the ICRP recommendations.

Codes of practice and other **guidance** assist designers and operators.

Facility design: radiation protection begins at the design stage. The process should include a risk assessment and an ALARA review.

Operations are carried out in accordance with standing orders or other written procedures.

Review and supervision is carried out at all levels of the process from the setting of standards down to operations on individual facilities.

The Radiation Protection Department of any establishment or facility is a service organization. Its roles include operation of a personnel monitoring system, surveillance of radiological conditions in the facility, provision of calibrated monitoring equipment, testing and accounting for radioactive sources, and maintenance of records of these activities.

Retention of records: There is a legal requirement for the retention of records for a period defined in the regulations.

Training is a key part of the process and should include both radiation protection and plant familiarization.

Appendix A: Relationship of units

Radiological quantity	Old unit	SI unit	Relationship between units	
Activity of a radioactive material	The curie (Ci)	The becquerel (Bq)		
	1 Ci = 3.7×10^{10} dis/s	1 Bq = 1 dis/s	1 Bq = 2.7×10^{-11} Ci	1 µCi = 37 kBq
		10^3 Bq = 1 kilobecquerel (kBq)	1 kBq = 2.7×10^{-8} Ci	1 mCi = 37 MBq
		10^6 Bq = 1 megabecquerel (MBq)	1 MBq = 2.7×10^{-5} Ci = 27 µCi	1 Ci = 37 GBq
		10^9 Bq = 1 gigabecquerel (GBq)	1 GBq = 27 mCi	
		10^{12} Bq = 1 terabecquerel (TBq)	1 TBq = 27 Ci	10^3 Ci = 37 TBq
		10^{15} Bq = 1 petabecquerel (PBq)	1 PBq = 27 kCi	10^6 Ci = 37 PBq
		10^{18} Bq = 1 exabecquerel (EBq)	1 EBq = 27 MCi	10^9 Ci = 37 EBq
Absorbed dose	The rad 1 rad = 0.01 J/kg	The gray (Gy) 1 Gy = 1 J/kg	1 µGy = 0.1 mrad 1 mGy = 100 mrad	1 mrad = 10 µGy 1 rad = 10 mGy
		1 Gy = 10^3 mGy = 10^6 µGy	1 Gy = 100 rad	100 rad = 1 Gy
Equivalent dose	The rem 1 rem = 1 rad × Q, where Q is the quality factor	The sievert (Sv) 1 Sv = 1 Gy × w_R, where w_R is the radiation weighting factor	1 µSv = 0.1 mrem 1 mSv = 100 mrem	1 mrem = 10 µSv 1 rem = 10 mSv
		1 Sv = 10^3 mSv = 10^6 µSv	1 Sv = 100 rem	100 rem = 1 Sv

Appendix B: Answers to numerical questions

CHAPTER 1

2. ${}^{4}_{2}\mathrm{He}$; two protons, two electrons, two neutrons: ${}^{23}_{11}\mathrm{Na}$; 11, 11, 12: ${}^{238}_{92}\mathrm{U}$; 92, 92, 146.
3. Masses are 1, 1/1840, 1; Charges are +1, −1, 0.

CHAPTER 2

2. (a) ${}^{234}_{90}\mathrm{Th}$; (b) ${}^{3}_{2}\mathrm{He}$; (c) ${}^{62}_{28}\mathrm{Ni}$
4. 2.3 min
5. (a) 5 MBq; (b) 0.75 MBq; (c) 1300 MBq; (d) 1 MBq
7. ${}^{59}_{27}\mathrm{Fe}$, ${}^{24}_{11}\mathrm{Na}$, ${}^{240}_{94}\mathrm{Pu}$

CHAPTER 3

5. 2.65×10^{7} n/(m^2 s).
6. 84 μSv.
7. 1.2 mSv.

CHAPTER 4

6. 2.5×10^{-4}, 10^{-3}.

CHAPTER 5

3. 2.20 mSv/year × 30 years = 66 mSv.

CHAPTER 6

3. 20 mSv, 500 mSv, 500 mSv
5. 40 mSv
6. 500 mSv, ~170 mSv, 2000 mSv.

CHAPTER 8

2. 10 h, 2.5 μSv/h
3. 225 μSv/h, 3 m
4. 6 m
5. 200 μSv/h

CHAPTER 9

1. 11.1 mSv
2. 313 Bq
4. 4 Bq/cm^2
5. 1.67×10^4 Bq/m^3

CHAPTER 10

2. 1970 Bq
4. Maximum β-energy 1.71 MeV, half-life 14.3 days, phosphorus-32.
6. 382 ± 3.4 counts/min.

CHAPTER 15

2. Approximately 21
6. 8.1×10^{-4}

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The Nuclear Energy Agency (NEA) within the Organisation for Economic Co-operation and Development (OECD) is a very good source of information on legislation and regulations. Further details can be obtained from its website at http://www.oecd-nea.org

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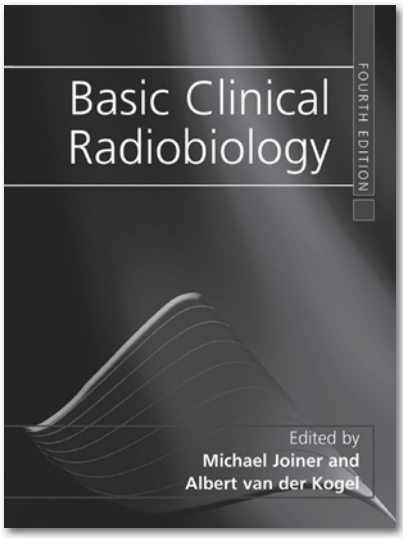

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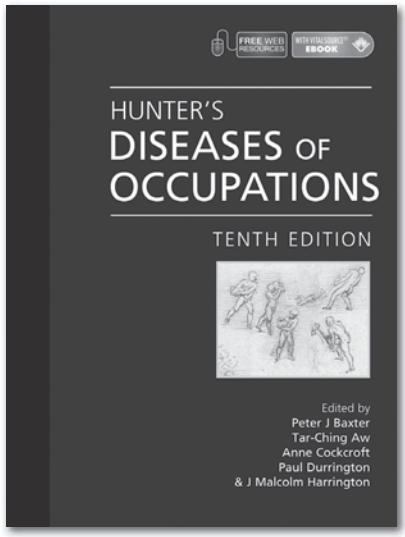

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