Radiation protection in nuclear facilities
Background

In 1991, the General Conference (GC) in its resolution RES/552 requested the Director General to prepare ‘a comprehensive proposal for education and training in both radiation protection and in nuclear safety’ for consideration by the following GC in 1992. In 1992, the proposal was made by the Secretariat and after considering this proposal the General Conference requested the Director General to prepare a report on a possible programme of activities on education and training in radiological protection and nuclear safety in its resolution RES1584.

In response to this request and as a first step, the Secretariat prepared a Standard Syllabus for the Postgraduate Educational Course in Radiation Protection. Subsequently, planning of specialised training courses and workshops in different areas of Standard Syllabus were also made. A similar approach was taken to develop basic professional training in nuclear safety. In January 1997, Programme Performance Assessment System (PPAS) recommended the preparation of a standard syllabus for nuclear safety based on Agency Safely Standard Series Documents and any other internationally accepted practices. A draft Standard Syllabus for Basic Professional Training Course in Nuclear Safety (BPTC) was prepared by a group of consultants in November 1997 and the syllabus was finalised in July 1998 in the second consultants meeting.

The Basic Professional Training Course on Nuclear Safety was offered for the first time at the end of 1999, in English, in Saclay, France, in cooperation with Institut National des Sciences et Techniques Nucléaires/Commissariat a l’Energie Atomique (INSTN/CEA). In 2000, the course was offered in Spanish, in Brazil to Latin American countries and, in English, as a national training course in Romania, with six and four weeks duration, respectively. In 2001, the course was offered at Argonne National Laboratory in the USA for participants from Asian countries. In 2001 and 2002, the course was offered in Saclay, France for participants from Europe. Since then the BPTC has been used all over the world and part of it has been translated into various languages. In particular, it is held on a regular basis in Korea for the Asian region and in Argentina for the Latin American region.

In 2015 the Basic Professional Training Course was updated to the current IAEA nuclear safety standards. The update includes a BPTC text book, BPTC e-book and 2 “train the trainers” packages, one package for a three month course and one package is for a one month course. The” train the trainers” packages include transparencies, questions and case studies to complement the BPTC.

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Editorial Note

The update and the review of the BPTC was completed with the collaboration of the ICJT Nuclear Training Centre, Jožef Stefan Institute, Slovenia and IAEA technical experts.
CONTENTS

1 INTRODUCTION ................................................................. 5
  1.1 Electromagnetic radiation .............................................. 5
  1.2 Ionizing radiation .......................................................... 5
  1.3 Natural and artificial sources of ionizing radiation .......... 7
  1.4 Development of radiation science ................................. 8

2 INTERACTION OF RADIATION WITH MATTER .......... 11
  2.1 Radiation, ionising radiation ........................................ 11
  2.2 Interaction of charged particles with matter ................. 12
  2.3 Interaction of gamma rays with matter ........................ 13
  2.4 Interaction of neutrons with matter .............................. 16
  2.5 Exercise ...................................................................... 18

3 RADIATION DETECTION ................................................. 19
  3.1 Physical basis of radiation detection ........................... 19
  3.2 Gas detectors .............................................................. 20
    Ionization chamber .......................................................... 24
    Proportional counter ........................................................ 24
    Geiger-Mueller counter .................................................... 24
  3.3 Scintillation detector .................................................... 24
  3.4 Thermoluminescent dosimeter (TLD) and optically
      stimulated luminescence dosimeter (OSLD) ............... 25
  3.5 Neutron detection ........................................................ 26
    BF\textsubscript{3} proportional detector .................................. 27
    Compensated ionization chamber ................................... 27
    Fission chamber .............................................................. 28
  3.6 Questions .................................................................... 28

4 DOSIMETRIC QUANTITIES ............................................. 29
  4.1 Absorbed dose (D) ...................................................... 29
    Dose rate for point sources of radiation ........................... 30
    Dose rate for other source geometries ............................. 31
  4.2 Equivalent dose (H) .................................................... 32
  4.3 Effective dose (E) ..................................................... 33
  4.4 Collective dose (E\textsubscript{c}) .................................. 34
  4.5 Exposure (X) ............................................................... 35
  4.6 Exercises .................................................................... 35

5 BIOLOGICAL EFFECTS OF RADIATION ....................... 37
  5.1 Basic cell biology ...................................................... 37
  5.2 Radiation effects on cells ............................................ 39
  5.3 Radiation effects on humans ...................................... 42
    Deterministic effects of radiation .................................. 42
    Stochastic effects of radiation ....................................... 44
  5.4 Questions .................................................................... 46

6 EXTERNAL RADIATION EXPOSURE ...................... 47
  6.1 Types of radiation exposure ........................................ 47
  6.2 Time, distance, shielding ............................................ 47
    Time ................................................................................ 48
    Distance .......................................................................... 48
Module II: Radiation protection in nuclear facilities

Shielding ................................................................. 48
6.3 Shielding from gamma radiation ......................... 49
6.4 Shielding from beta radiation .............................. 50
6.5 Shielding from neutron radiation ........................ 51
6.6 Exercises .......................................................... 51

7 INTERNAL RADIATION EXPOSURE ....................... 53
7.1 The pathways of radionuclides into the body .......... 53
7.2 Effective half-life, committed effective dose .......... 54
7.3 Dose coefficient, derived air concentration .......... 55
7.4 Exposure to radioactive noble gases ................. 57
7.5 Protection against internal contamination .......... 58
7.6 Protective clothing and equipment ........................ 58
   Respiratory protective equipment ......................... 60
   Fit test ............................................................. 63
7.7 Exercises .......................................................... 64

8 RADIATION PROTECTION REGULATIONS ............. 65
8.1 The aim of radiation protection ......................... 65
8.2 International recommendations and standards ........ 66
   ICRP recommendations ......................................... 66
   IAEA Safety standards .......................................... 67
8.3 Exercises and questions ...................................... 72

9 RADIATION PROTECTION IN NUCLEAR INSTALLATIONS ........................................ 73
9.1 Aspects of radiation protection in the nuclear fuel cycle73
9.2 Important radionuclides for radiation protection ....... 74
   Natural radionuclides ............................................ 75
   Fission products .................................................. 76
   Activation and corrosion products ......................... 77
   Hot particles ...................................................... 78
9.3 Questions .......................................................... 79

10 ENVIRONMENTAL MONITORING ....................... 80
10.1 Need for monitoring nuclear facilities ................. 80
10.2 Exposure pathways to population ....................... 81
10.3 Objectives of monitoring .................................... 83
10.4 Programmes for environmental monitoring .......... 84
10.5 Assessment of doses to members of the public ....... 85
10.6 Exposures of population from various stages of fuel cycle......................................................... 87
10.7 Natural background ............................................ 89
10.8 Monitoring in emergency exposure situations ........ 90
   Objectives of emergency monitoring ..................... 91
   Source monitoring during the emergency ............... 91
   Environmental monitoring during the emergency ..... 92
   Personal monitoring ............................................. 94
10.9 Questions .......................................................... 95

11 REFERENCES ...................................................... 96
1 INTRODUCTION

Learning objectives
After completing this chapter, the trainee will be able to:
1. Describe the concept of radiation.
2. Describe the spectrum of electromagnetic radiation.
3. Define ionizing radiation.
4. Identify the types of ionizing radiation.
5. Explain the main properties of various types of ionizing radiation.
6. Explain the difference between natural and artificial sources of ionizing radiation.
7. Describe the discovery of X-radiation and the discovery of radioactivity.
8. List the harmful health effects observed in the early period of use of ionizing radiation.
9. Outline the historical development of recommendations on radiation protection.
10. Describe the role of the ICRU and ICRP.

1.1 Electromagnetic radiation

Radiation is a physical phenomenon where a body gives off energy and this energy travels through space until absorbed by another body. Typical examples of radiation which can be perceived by the human senses are sound and light. Sound is a mechanical wave which can only spread through matter (and is of no further interest to us), unlike light, which travels most freely through completely empty space (a vacuum).

The energy carried by electromagnetic radiation through space cannot be broken down into arbitrarily small parts. As it turns out, there is a minimum energy packet of electromagnetic radiation which cannot be further separated into smaller amounts of energy. Such a packet of electromagnetic radiation is called a photon. The energy of a photon is inversely proportional to the wavelength of the radiation. The energy of photons of visible light is around 2 eV, while photons contained in radiation with a wavelength shorter or longer than visible light have correspondingly higher or lower energy, respectively.

1.2 Ionizing radiation

Of interest particularly is radiation which has a wavelength shorter than 100 nm or an energy higher than 12 eV. Since this energy is greater than the binding energy of electrons in the atom, electromagnetic radiation with a wavelength below 100 nm is capable of removing electrons from the atoms of the matter it is moving
through. Radiation passing through matter with a large enough photon energy produces electron-ion pairs or ion pairs. This phenomenon is called ionization and the associated radiation is ionizing radiation. Within the electromagnetic spectrum (Figure 1.1), ionizing radiation includes ultraviolet (partly), X and gamma radiation. Although these types of radiation differ from each other in energy and origin, the mechanisms of their interaction with matter are practically identical and will from now on be collectively considered as gamma radiation, because this is the type of electromagnetic radiation we will most often deal with.

The radiation emitted by mobile phones or base stations has a wavelength of above 10 cm. Since the photon energy of such radiation is a million times smaller than the ionization energy, mobile phones clearly do not emit ionizing radiation.

In addition to electromagnetic radiation with a large enough photon energy, other forms of ionizing radiation include atom-sized particles or of smaller mass, produced from radioactive decay, nuclear fission and in accelerators. Their energies are of the order of magnitude of MeV, which means they strongly ionize matter. It should be noted that (thermal) neutrons with an energy of a few hundredths of and eV or less also constitute ionizing radiation (on account of the nuclear reactions they cause).
Table 1.1: Characteristics of radiation.

<table>
<thead>
<tr>
<th>name (symbol)</th>
<th>characteristic</th>
<th>mass</th>
<th>charge</th>
<th>typical sources</th>
<th>penetration depth</th>
</tr>
</thead>
<tbody>
<tr>
<td>alpha (α)</td>
<td>$^4$He nuclei</td>
<td>$\approx 4$ u</td>
<td>$+2e_0$</td>
<td>heavy nuclei</td>
<td>air: ~ cm, sheet of paper</td>
</tr>
<tr>
<td>beta (β⁻)</td>
<td>electrons</td>
<td>$0.00055$ u</td>
<td>$-e_0$</td>
<td>fission and activation prod.</td>
<td>air: ~ m, Al: ~ mm</td>
</tr>
<tr>
<td>beta (β⁺)</td>
<td>positrons</td>
<td>$0.00055$ u</td>
<td>$+e_0$</td>
<td>fission and activation prod.</td>
<td>air: ~ m, Al: ~ mm</td>
</tr>
<tr>
<td>gamma (γ)</td>
<td>EM radiation</td>
<td>-</td>
<td>0</td>
<td>almost all radionuclides</td>
<td>Pb: ~ dm, water: ~ m</td>
</tr>
<tr>
<td>neutron (n)</td>
<td>nucleon</td>
<td>$\approx 1$ u</td>
<td>0</td>
<td>reactor</td>
<td>paraffin: ~ m, water: ~ 10 m</td>
</tr>
</tbody>
</table>

In the context of radiation protection, our interest will focus almost exclusively on ionizing radiation. For simplicity’s sake, we will often drop the adjective “ionizing” and refer to radiation only. The most important types of radiation encountered in nuclear technology and their main characteristics are given in Table 1.1.

### 1.3 Natural and artificial sources of ionizing radiation

Radioactivity and ionizing radiation have been present on Earth since its very creation. All sources of radiation that occur naturally without their origin or intensity being influenced by humans in any way are called **natural sources of radiation**. They are grouped into three families:

- very long-lived radionuclides and their progeny,
- cosmic radiation,
- cosmogenic radionuclides.

Very long-lived radionuclides have such a long half-life that they have not decayed away since the beginning of the Earth. They include three nuclides from the end of the periodic table ($^{232}$Th, $t_{1/2} = 1.4 \times 10^{10}$ years, $^{235}$U, $t_{1/2} = 7.10^8$ years, $^{238}$U, $t_{1/2} = 4.5 \times 10^9$ years) and their radioactive daughter products which form radioactive chains ending in stable lead. These daughter products have short enough half-lives to have decayed long since the origin of the Earth; however, being continuously produced by the decay of the long-lived chain originator, radioactive equilibrium between them is maintained. Within these natural radioactive chains, nuclides of special importance for radiation protection include $^{226}$Ra ($t_{1/2} = 1600$ years) and its daughter product, the noble gas $^{222}$Rn ($t_{1/2} = 3.8$ days).

There are also a few very long-lived nuclides which are not members of the natural radioactive chains from the end of the periodic table. The key nuclide of this kind is $^{40}$K ($t_{1/2} = 1.3 \times 10^9$ years).

**Cosmic radiation** is ionizing radiation which comes from space. Its source is so far unknown. It largely consists of high-energy (~GeV)
protons that trigger nuclear reactions in the upper layers of the atmosphere, producing other ionizing particles (gamma rays, neutrons) which reach the Earth’s surface. Cosmic radiation makes a significant contribution to natural background radiation or natural human exposure to radiation. The intensity of cosmic radiation increases with altitude; during a transatlantic flight we will receive an additional dose comparable to a weekly dose of natural radiation on ground.

Cosmic radiation is also the cause of the formation of **cosmogenic radionuclides**. Though their half-lives are relatively short, they are continuously formed by nuclear reactions which are triggered by cosmic radiation in the atmosphere. The most important cosmogenic nuclides are: $^3\text{H}$ ($t_{1/2} = 12.3$ years), $^7\text{Be}$ ($t_{1/2} = 53$ days) and $^{14}\text{C}$ ($t_{1/2} = 5760$ years).

**Artificial sources of radiation** are sources made by man (i.e. anthropogenic). They include:
- X-ray devices,
- particle accelerators,
- artificial radionuclides,
- nuclear reactors,
- materials with a technologically enhanced concentration of natural radionuclides (TENORM),

In nuclear technology, the major sources of radiation are nuclear reactors, which are simultaneously sources of artificial radionuclides.

### 1.4 Development of radiation science

Radiation has been part of the natural world ever since the birth of the universe, just as humanity has always been exposed to various forms of natural radiation throughout its evolution. We were not aware of ionizing radiation, until two major discoveries were made in late 19th century. First, on 8 November 1895, Wilhelm Conrad Röntgen discovered penetrating rays which he called X-rays (nowadays they are also called roentgen rays). Only a few months later (1 March 1896), Henry Becquerel discovered a phenomenon which was later named *radioactivity* by Marie Curie.

Soon after both discoveries, X-rays in particular were found to have practical value (e.g. to take pictures of the inside of the human body, see Figure 1.2), so X-ray devices and radioactive substances started being used in physics, chemistry and medicine. In those pioneering times, users were not aware that excessive exposure to radiation has serious biological consequences. Nor did they have any accurate instruments for measuring the intensity of radiation.

The earliest use of radiation was very soon followed by cases of damage from intensive exposure to sources of radiation. In 1896, it was found that exposure to X-rays may cause *erythema* (abnormal
skin redness), oedema (water accumulation in tissues) and epilation (hair loss). In this period, X-ray tubes were calibrated simply on the basis of the erythema caused on a hand exposed to X-radiation by the operator.

Soon after, carcinogenic or cancer-inducing effects of radiation were also observed. By 1911, 94 cases of tumours caused by X-radiation were reported, 50 of them among radiologists (doctors specialized in the use of radiation for medical purposes). In 1922, an estimate was published that 100 radiologists had died from radiation-induced cancer. Somewhat later, it was discovered that leukaemia (also called cancer of the blood or bone marrow) is much more frequent in radiologists than other physicians. Marie Curie who worked with radioactive materials died of leukaemia in 1934 aged 67 years.
Figure 1.2: An X-ray of Mrs Röntgen’s hand, taken on 22 Dec. 1895, less than two months after the discovery of X-rays.

Along with its obvious benefits, the use of radiation was thus accompanied by incontrovertible proof of its harmful effects. Relatively early on, this gave rise to the first recommendations for reducing excessive exposure to radiation, which eventually developed into a special scientific discipline – radiation protection.

The first limits on exposure to radiation were intended to prevent acute effects such as skin lesions which occur after high exposure to radiation. Later, limits were introduced to prevent more insidious effects causing harm in the long term, such as cancer. Greater cancer morbidity was first noted in miners working in mines containing uranium ores and in female workers who used paintbrushes to apply radium paint to dials of clocks and instruments from 1915 to 1930. To be able to paint sharper numbers on the dial, they sharpened the tip of the paintbrush with their lips, thereby taking in considerable quantities of radium. However, the key group which allowed the long-term effects of radiation to be quantitatively determined with greater precision were the survivors of the atomic bombs at Hiroshima and Nagasaki. It should be noted, however, that all the data obtained in this way is based on individuals who were exposed to relatively high radiation in a short amount of time. The greatest challenge of radiation protection is how to use the data from high exposure in evaluating the risk posed by accumulated low level exposure to radiation, which is most commonly the case in occupationally exposed workers.

In the early period, there was a fair amount of confusion about the quantities and units with which to measure exposure to radiation. For this reason, the International Commission on Radiation Units & Measurements (ICRU) was established in 1924. The first dosimetric quantity, exposure, and its unit, the roentgen (R), were defined in 1928.

1928 also saw the establishment of the International Commission on Radiological Protection (ICRP) which collects scientific data on the effects of radiation and based on such data issues recommendations on radiation protection.

Thus, in contrast to common belief, the health effects of ionizing radiation are now very extensively studied and known, probably much better than can be claimed for most other environmental substances harmful to health.
2 INTERACTION OF RADIATION WITH MATTER

Learning objectives
After completing this chapter, the trainee will be able to:
1. Explain the interaction of charged particles with matter.
2. Compare the range and shape of paths in matter for various charged particles.
3. Describe the three most important types of interaction of gamma radiation with matter.
4. Define the half-value layer and the tenth-value layer.
5. Use the equation for transmitted gamma ray flux as a function of substance thickness and draw the relevant function.
6. Describe the interaction of neutrons with matter.
7. Describe the terms “fast neutron” and “thermal neutron”.
8. Describe the typical energy dependence of the neutron capture cross section.

2.1 Radiation, ionising radiation

The term radiation is understood to include various fast particles (e.g. electrons, alpha particles, neutrons) and electromagnetic radiation. As regards the latter, nuclear technology is particularly concerned with gamma and X-radiation. Energy packets of electromagnetic radiation are called photons and they are treated in a similar way as other (mass) particles.

Radiation particles have considerable energy, usually of the order of magnitude of keV to MeV. When radiation passes through matter, the particles collide with atoms or their nuclei and react with them. This phenomenon is called the interaction of radiation with matter. Radiation loses its energy by passing it on to atoms or nuclei in matter. This energy transfer can happen by a number of different physical processes which depend on the type and energy of the radiation. As a rule, if the energy transferred is large enough, these processes finish by stripping electrons away from the atom. An atom which loses an electron of course becomes an ion. In this way, ions and free electrons form along the path of charged particles. The ion-electron pair is called the ion pair and the formation of ion pairs is termed ionization. Hence we speak of ionizing radiation.

In the interaction of radiation with matter, the main dividing line is between the interaction of charged particles and the interaction of electrically neutral particles with matter. Electrically charged particles (alpha and beta particles, fission fragments) interact with atoms or electrons and nuclei by means of electrostatic force and directly cause ionization in matter. In this case we speak of direct ionization.
Electrically neutral particles (neutrons and gamma rays), on the other hand, interact with atomic nuclei or electrons in various processes causing the formation of charged particles (electrons and/or ions) with a kinetic energy large enough for these secondary charged particles to ionize matter. This is called indirect or secondary ionization.

### 2.2 Interaction of charged particles with matter

The most important charged particles in nuclear technology are alpha particles, beta particles and fission fragments. Alpha particles and fission fragments are considered heavy charged particles and beta particles light charged particles. Electrically charged particles interact with atoms or electrons and nuclei by means of electrostatic force and cause direct ionization.

By forming ion pairs, the charged particles lose energy and slow down. The intensity of interaction between charged particles and matter is described by the following quantities:

- **Specific ionization** is the number of ion pairs per unit of path length;
- **Stopping power** is the loss of energy per unit of path length.

Another important quantity is:

- **Range** – maximum depth a particle penetrates through matter.

Heavy charged particles (alpha particles, fission fragments) are characterized by large specific ionization and stopping power and a small range. The path of heavy particles is linear since they are considerably heavier than electrons in matter. The range of an alpha particle with 2 MeV energy is around 1 cm in air but only a few µm in denser substances.

![Figure 2.1: Path of an electron in matter.](image)
Light charged particles (beta particles or electrons and positrons) have much smaller specific ionization and stopping power than heavy charged particles. Since their mass is identical to the mass of electrons, they are deflected on colliding with electrons in atoms to the same degree as the electrons. This is why their path follows a series of angles or zig-zags. Their range is longer than for heavy charged particles; a beta particle of 2 MeV energy has a range of around 8 m in air and a few mm in metals.

### 2.3 Interaction of gamma rays with matter

Gamma rays are photons or packets (quanta) of electromagnetic radiation with a very short wavelength. X-rays are similar in this respect, except that γ rays originate from changes in atomic nuclei, whereas X-rays originate from the electron cloud (and not the nucleus) of an atom and generally have less energy than gamma rays. All other properties of gamma and X radiation are the same, so the interaction of both types of electromagnetic radiation with matter will be discussed together.

Photons move in a straight line at constant speed (the speed of light). As they travel through matter, they interact with atomic electrons and in some cases also with atomic nuclei. The most important interactions between gamma photons and atoms are:
- the photoelectric effect,
- Compton scattering,
- pair production.

**Figure 2.2:** The photoelectric effect.

The **photoelectric effect** leads to complete photon absorption. The photoelectric effect involves a photon hitting one of the bound electrons in an atom and passing all its energy to this electron. The photon disappears as the electron is emitted by the atom with a kinetic energy equal to the photon’s energy minus the electron’s binding energy. This binding energy is usually considerably smaller than the
gamma ray energy, so the knocked-out electron has considerable kinetic energy and travels on through matter like a beta particle would.

**Compton scattering** is a process in which a photon hits an electron in an atom. Part of the photon’s energy is received by the struck electron which consequently starts moving through matter like a beta particle, while the remaining energy is emitted in the form of a photon. Its energy is clearly smaller than that of the original photon, since the overall energy is conserved. Momentum is also conserved, which is why the electron and new photon fly off at defined angles.

![Figure 2.3: Compton scattering.](image)

**Pair production** occurs when a high energy photon approaches an atomic nucleus and the strong electric field of the nucleus creates a positron and an electron pair which fly off in opposite directions while the photon disappears. The production of the electron-positron pair requires 1.02 MeV, and the remaining energy is shared among the electron and positron. Pair production is relevant only to photons with high energy (>> 1.02 MeV).

![Figure 2.4: Pair production.](image)

Unlike charged particles, which continuously lose energy as they
travel through matter, photons interact with atoms more rarely, but when they do they lose all their energy (absorption) or a considerable portion of it (scattering) at once. Consequently, the range of photons cannot be defined like for charged particles; we can only predict the probability of a photon scattering or being absorbed as it traverses a certain thickness of matter. This means that the interaction of gamma rays with matter is a random phenomenon. The probability of interaction per unit path length is given by the linear attenuation coefficient, \( \mu \).

**Figure 2.5:** Exponential attenuation of the transmitted flux of photons passing through matter.

In mathematical terms, the process of photons travelling through matter may be compared to radioactive decay which is likewise a random process. Just as the number of radioactive nuclei diminishes exponentially with time, the flux of monoenergetic photons, \( j \), which travel a certain distance without scattering or being absorbed also reduces exponentially. In this context, the **half-value layer**, \( d_{1/2} \), is defined as the thickness of a substance at which the flux of photons is reduced to half its value (Figure 2.5).

The flux of transmitted photons passing through matter is described by the equation:

\[
j = j_0 \cdot 2^{-\frac{d}{d_{1/2}}}.
\]

In addition to the half-value layer we also refer to the tenth-value layer, \( d_{1/10} \), the thickness of a substance at which the flux of photons is reduced by a factor of 10; after passing two tenth-value layers by a factor of 100, after three tenth-value layers by a factor of 1000, etc.
The exponential law of attenuation can be expressed through the tenth-value thickness:

\[ j = f_0 \frac{\rho}{d_{1/10}}. \]

The half-value layer and tenth-value layer are related as follows:

\[ d_{1/10} = 3.3 d_{1/2}. \]

### 2.4 Interaction of neutrons with matter

Neutrons are electrically neutral particles, so there is no electric interaction with electrons or atomic nuclei. They interact with atomic nuclei only by **nuclear reactions** via the nuclear force, which, however, has a very short range. Since the nucleus takes up a very small part of the atom’s volume (the radius of the nucleus is ten thousand times smaller than the radius of the atom) neutrons travel through matter as through almost empty space. Only now and then neutrons happen to collide with atomic nuclei, when the neutron either bounces off the nucleus (neutron **scattering**) or is **absorbed** by it, causing a nuclear change. Owing to the random nature of neutron interaction with matter, the intensity (flux) of neutrons **attenuates exponentially** with the thickness of substance.

Neutrons released in nuclear fission or other nuclear reactions have kinetic energy of the order of magnitude of MeV; they are called **fast neutrons**. On colliding with the nuclei (scattering) they lose some of their kinetic energy. This energy loss is the greatest when scattering on light nuclei, which are present especially in the **moderator**. After scattering, neutrons continue to travel through matter, but with less kinetic energy than before colliding with a nucleus. This is why neutrons are said to **slow down** by scattering. Consecutive scattering gradually reduces the energy of neutrons until on average it is equal to the kinetic energy of thermally moving atomic nuclei of matter. Such neutrons are called **thermal neutrons**.

In contrast to scattering where a neutron slows down, **absorption** involves a nucleus capturing a neutron. The most common absorption reactions include radiative capture or the (n, γ) reaction, capture reactions with emission of other particles such as (n, α) and (n, p) reactions, and fission or (n, f) reaction.

In radiative capture or capture with emission of other particles the neutron does not reappear after the reaction, hence these reactions reduce the number of neutrons in matter. Conversely, a fission reaction increases the number of neutrons.
The radiative capture reaction is the most frequent neutron absorption reaction. For most nuclides, it may only occur with thermal neutrons. In this reaction, a nucleus absorbs a neutron and the resulting excited nucleus de-excites by emitting a gamma ray. With certain nuclides, thermal neutrons can also induce (n, α) and (n, f) reactions, whereas as a rule all other reactions are threshold reactions. This means that neutrons require kinetic energy high enough (typically of the order of magnitude of MeV) to induce such reactions.

The absorption cross section \( \sigma_a \) has three distinct regions depending on neutron energy:
- \( 1/\nu \) region,
- resonance region,
- high energy region.

**Figure 2.6:** Cross section for radiative capture in \(^{58}\text{Fe}\).

As two examples, Figures 2.6 and 2.7 show the cross sections for radiative capture in \(^{58}\text{Fe}\) and for fission of \(^{235}\text{U}\). Both axes in the figures have logarithmic scales.

**Figure 2.7:** Cross section for \(^{235}\text{U}\) fission.

For the lowest energies, the absorption cross section is inversely
proportional to neutron speed, hence this region is called the $1/v$ region.

In the resonance region, the cross section value oscillates strongly. The cross section value is extremely large for the neutron energies which on collision give the nucleus exactly the energy it needs to be excited into a particular excited state. We say that the neutron is in resonance with the nucleus. For other energies, the cross section values between individual resonances are much lower. The resonance region usually corresponds to neutron energies from 1 eV to 100 keV. Such neutrons are called epithermal neutrons.

For high energies (above 100 keV), individual resonances overlap to such a degree that the end result is a once again smoother line of cross section values which slowly falls as the energy increases. Neutrons in this energy region are classed as fast neutrons.

### 2.5 Exercise

1. The half-value layer of a substance for photons of 1.5 MeV energy is 1 cm. By how much is the photon flux reduced on passing through 5 cm of this substance? And on passing through 10 cm of the substance?
3 RADIATION DETECTION

Learning objectives

After completing this chapter, the trainee will be able to:

1. Explain the terms direct and indirect ionising radiation and give relevant examples of such radiation.
2. Describe the basic principle of detecting radiation with gas detectors.
3. Explain and draw the current-voltage characteristic of gas detectors.
4. Describe how an ionization chamber works.
5. Describe how a proportional detector works.
7. Describe how a scintillation detector works.
8. Describe how TL and OSL dosimeters work.
9. Describe neutron detection with a BF3 proportional detector.
10. Describe neutron detection with a compensated ionization chamber.

3.1 Physical basis of radiation detection

We are unable to detect ionizing radiation with our senses, but we can detect and measure it with instruments, which are based on the mechanisms by which radiation interacts with matter. The consequences of the interaction of radiation with matter that can be exploited for detection are:

- the production of free electrons and ions in matter (ionization),
- the production of excited atoms,
- the heating of matter,
- the occurrence of microscopic damage to matter,
- nuclear reactions in matter,
- bremsstrahlung (electromagnetic radiation which results from the deceleration of charged – in particular β – particles in matter).

The effect most frequently exploited for detection is ionization.

Regarding ionizing radiation, we distinguish two types of particles; those that ionize matter directly and those that cause ionization indirectly. The first group includes charged particles (e.g. electrons or beta particles, protons, alpha particles and heavy ions) and the second includes photons (gamma particles and X-radiation) and neutrons. As regards photons, ionization is mostly caused by electrons resulting from the photoelectric effect, Compton scattering or pair production. As for neutrons, ionization is caused by the charged particles resulting from nuclear reactions, e.g. α particles produced by the (n, α) reaction.

The device in which ionization is observed is called a detector. The volume of matter used to collect the resulting charges is called the
active volume of the detector and must be large enough to absorb a substantial part of the energy of the radiation. This is why alpha particle detectors can be much smaller than photon or neutron detectors. Based on the type of matter undergoing ionization, the following main types of detectors are distinguished:

- gas detectors,
- scintillation detectors,
- semiconductor detectors,
- other detectors.

In some detectors, the number of pulses measured is related to the energy absorbed in the detector. Such a detector can be calibrated to show the absorbed dose, which is why it is called a dosimeter.

### 3.2 Gas detectors

In principle, a gas detector is a condenser inside which radiation ionizes a gas. Ion pairs are formed; negative particles (electrons) accelerate towards the positive electrode (anode) and positive ones (ions) towards the negative electrode (cathode). This produces an electric current which is then additionally amplified and registered (Figure 3.1).

![Figure 3.1: The principle of gas detector operation.](image)

The most common gas detector geometry is the coaxial detector (Figure 3.2). Such a detector is shaped like a long cylindrical tube which contains gas. A thin wire runs along its axis and is electrically insulated from the tube wall. There is positive voltage on the wire, hence the anode, whereas the wall is the cathode. Gamma radiation interacts primarily with the tube wall, via photoelectric effect, Compton scattering, or pair production. The secondary electrons from these interactions travel inside the detector and ionize the gas. However, if the radiation measured contains beta or alpha particles, the tube should be designed to allow these particles to enter the
detector. This can be achieved with a very thin membrane at one side of the tube, which is called the detector window.

**Figure 3.2**: Coaxial gas detector.

If a gas detector is exposed to a constant flux of ionizing particles and the voltage, $U$, is changed, the current produced, $I$, also changes. The dependence of current on voltage is called the current-voltage characteristic of the detector (Figure 3.3); it may be divided into 6 distinct regions marked with Roman numerals from I to VI.

**Figure 3.3**: Current-voltage characteristic of a gas detector.

In the first region, at low voltage, the electric field in the detector is weak and the ion pairs created often meet an oppositely charged particle and recombine into electrically neutral atoms or molecules. If the voltage, $U$, is higher, the movement of ions towards the electrodes
is more direct, the probability of recombination falls and the current, $I$, rises. Nevertheless, in this region some of the created ion pairs still recombine before reaching the electrodes, so the region is called the **recombination region**. This region is not normally used for radiation detection.

In region II the voltage is high enough for all of the released ion pairs to reach the electrodes and as a result contribute to the current, $I$. This means there is no recombination. Any additional increase in the voltage does not increase the current, $I$. This constant current is directly proportional to the number of ion pairs produced or to the energy deposited in the detector by radiation. This voltage region is called the **ionization chamber region**. The eponymous detector operates within this region.

At higher voltage, $U$, the electric field in the counter is amplified enough for the primary electrons released in the counter by radiation to collide with gas molecules and ionize them. This produces new (secondary) electrons which are also able to ionize further gas molecules. This leads to an electron avalanche which reaches the wire anode (Figure 3.4). In this way, each primary electron sets off an avalanche of secondary electrons, which greatly increases the current, $I$ (region III in Figure 3.3). The resulting current changes with voltage, but at a given voltage the factor of primary charge multiplication is constant, which means that the resulting current is directly proportional to the number of primary ion pairs. This voltage region is called the **proportional region**, in which the proportional counter operates.

![Electron avalanche](image.png)

**Figure 3.4:** Electron avalanche.

At voltages above the proportional region (region IV in Figure 8.3) the avalanche multiplication increases the primary charge even more than in the proportional region, but the multiplication factor is no longer independent of the number of primary charges. Hence this region is called the region of limited proportionality and is not normally used for radiation detection.

At an even higher voltage, the electric field is so strong that avalanches of electrons excite the internal electrons of gas atoms. The ultraviolet rays produced by the transition of these atoms into the ground state have enough energy to ionize other atoms throughout the
volume of the detector (Figure 8.5). This creates a very strong current; however, it is independent of the number of primary ion pairs. The current or pulse height does not change materially if the voltage on the tube changes. This region, marked V in Figure 3.3, is called the **Geiger-Mueller plateau**. The eponymous counter can only be used to detect the presence of radiation and not to measure its energy.

![Avalanche multiplication in a Geiger-Mueller detector.](image)

**Figure 3.5:** Avalanche multiplication in a Geiger-Mueller detector.

If the voltage is higher than the Geiger-Mueller plateau (region V in Figure 3.3), the gas in the detector is ionized just by virtue of the strong electric field and collisions between molecules. Even with no radiation present, this will create a strong current through the detector. This region is called the **breakdown region** and usually destroys the detector.

The dependence of the current-voltage characteristic of a detector on the number of primary ion pairs (i.e. the energy deposited in the detector by a particle) is shown in Figure 3.3 by the dotted curve which represents the current-voltage characteristic for an increased number of primary ion pairs released by one particle. A difference is seen only at low voltages, in the operational region of the ionization chamber and proportional counter. In these two regions the difference is proportional to the difference in energy deposited by different particles.

![Pulse and current mode of operation.](image)

**Figure 3.6:** Pulse and current mode of operation.

The electric signal from a detector can be measured in two ways. In **pulse mode** operation, we measure one event at a time, i.e. we
measure the change in detector voltage (pulse). The pulse height is proportional to the charge. In the current mode, we measure the average charge produced in the detector in a certain period of time, i.e. the electric current through the detector (Figure 3.6).

**Ionization chamber**

The ionization chamber usually operates in the current mode at voltages between 100 and 300 V. The filling gas is normally air. Since the electric current produced is relatively weak (around $10^{-11}$ A), it needs considerable amplification (around ten thousand-fold), which should be linear. The ionization chamber is more suitable for measuring strong radiation than for detecting individual events. The uses of ionization chambers include measurement of the absorbed dose and reactor power.

**Proportional counter**

Proportional counters operate at higher voltages than ionization chambers, typically around 300 to 700 V. The output signal is directly proportional to the energy deposited by incoming radiation. Due to avalanche multiplication the electrical signals are about a thousand times stronger than in ionization chambers. The filling gas is usually argon, xenon or methane. Proportional counters operate almost exclusively in the pulse mode.

**Geiger-Mueller counter**

Geiger-Mueller counters are filled with neon or argon and operate at voltages around 1000 to 1200 V. Its typical feature is that the primary ionization is multiplied by a factor from $10^4$ to $10^8$ and the resulting electric signal is very strong (of the order of magnitude of 0.1 V) but independent of the number of primary ion pairs. This means a Geiger-Mueller counter cannot be used to measure radiation energy; it can only be used to register individual ionization events. The signal collection time or electron avalanche quenching time is a few hundred microseconds. During the moment of primary ionization and the moment of avalanche quenching the counter cannot count, so this time is called the counter’s dead time, $\tau$. Dead time is important at high count rates or high radiation dose rates.

### 3.3 Scintillation detector

Some crystalline substances and organic materials (called scintillators) are luminescent, which means they emit light after absorbing radiation. The best-known example is a CRT TV screen which emits visible light when hit by an electron beam from the cathode tube. This effect is exploited by scintillation detectors. They consist of a scintillator which is transparent to its own light, a photomultiplier and supporting electronics (Figure 3.7). The most frequently used scintillator is a sodium iodine crystal doped with thallium, denoted by NaI(Tl).
Some of the energy of the absorbed radiation is emitted by the scintillator as electromagnetic radiation or light. The more incoming radiation is absorbed in the scintillator, the stronger the flash of light that is produced. The resulting light flash is guided to the photomultiplier where it is converted into an electric signal; its amplification gives us a strong electric pulse which is proportional to the energy of the light flash, i.e. to the energy deposited in the scintillator by incoming radiation.

![Scintillation detector](image)

**Figure 3.7:** Scintillation detector.

A photomultiplier changes light pulses into electric pulses. The output electric signal is directly proportional to the energy of incoming radiation absorbed in the scintillator. A scintillation counter can therefore measure the energy of incoming radiation. Its second advantage is that the pulse develops in an extremely short time (µs), so it can also be used in a strong radiation field. Finally, the scintillation crystal absorbs gamma rays much more effectively than the thin gas in gas detectors, so it is used chiefly to measure gamma radiation.

### 3.4 Thermoluminescent dosimeter (TLD) and optically stimulated luminescence dosimeter (OSLD)

To measure the quantities of radiation a human being has been exposed to we use **personal dosimeters**. They measure the dose received or the energy absorbed per unit mass of matter. In the past, a photographic emulsion was used for this purpose, but today we most frequently use a **thermoluminescent dosimeter (TLD)**, an **optically stimulated luminescence dosimeter (OSLD)** or an electronic dosimeter.

In thermoluminescent substances (e.g. CaF$_2$, LiF), radiation excites electrons to higher energy levels just like in scintillators. However, these electrons are not immediately restored to the ground state (emitting light in the process) as in scintillators but remain captured in what we call traps. The direct transition of an electron from a trap into the ground state is not possible. This only happens when additional
energy is supplied to them, which is done by heating. Luminescence occurs when a photon is emitted during the transition into the ground state (Figure 3.8).

Some dosimeters of more recent date are based on the optically stimulated luminescence (OSL) effect. The principle of operation is similar to the TLD, except that electron recovery from the trap to the ground state (and light emission) is not stimulated by heating but by light of selected wavelength. Al₂O₃ crystals can be used in OSL dosimeters; their luminescence is stimulated with a green light laser and the emitted light is blue.

![Figure 3.8: The principle of thermoluminescent dosimeter (TLD) operation.](image)

TL and OSL detectors are passive; just a tablet with luminescent material is irradiated and a reading is taken subsequently. After reading and “erasing” (heating), the tablet is reused. Depending on its composition, TLDs are used to establish the dose of X-rays, γ rays, β rays and neutrons. OSL dosimeters enable multiple readings.

### 3.5 Neutron detection

Like photons, neutrons belong to the family of indirectly ionizing particles. An important distinction is that photons react to a lesser or greater extent with virtually all substances to form charged particles, whereas neutrons form charged particles only in a relatively small number of nuclear reactions.

To detect slow, i.e. thermal neutrons, the following nuclear reactions are used:

\[
\begin{align*}
10^6 \text{B} (n, \alpha) & 7^\text{Li} \\
6^6 \text{Li} (n, \alpha) & 3^\text{H} \\
3^3 \text{He} (n, p) & 3^\text{H}
\end{align*}
\]
A substance containing the atoms of one of the targets in the above reactions is added to a standard gas counter (e.g. ionization chamber or proportional counter). Most commonly this substance is boron, added either in gaseous form (e.g. as BF$_3$ gas mixed with argon in the counter) or by lining the interior walls of the counter with a thin layer of a boron compound (e.g. boron carbide). On their way through the detector, neutrons collide with $^{10}$B nuclei, producing two positively charged nuclei of lithium and helium, which share the released reaction energy (2.79 MeV) among themselves. In most cases (94%) $^7$Li remains in an excited state, which lowers the kinetic energy of the products to 2.31 MeV, but even this is easily detected. The resulting electrical particles ionize the gas in the counter and this is registered in the usual way.

**BF$_3$ proportional detector**

As the name tells us, this is a proportional detector containing BF$_3$ filling gas. This is the most common neutron detector. It operates in the pulse mode but is not used to measure the energy of neutrons. Instead, this mode enables a clear distinction between the signal of an absorbed neutron and the signal of an absorbed γ ray (discrimination).

**Compensated ionization chamber**

This neutron detector effectively distinguishes neutron radiation signals from gamma radiation signals. In principle, it involves two chambers: the walls of the first are lined with boron to make it sensitive to both neutrons and gamma rays, whereas the other is a standard chamber and hence sensitive to gamma rays only. The current in the first chamber results from both neutron and gamma radiation, while that in the second chamber from gamma radiation only, so the output signal (if compensation is properly performed) is equal to the difference between the currents in the first and second chamber and therefore proportional to the neutron flux only (Figure 3.9). The part of the compensated ionization chamber which is only sensitive to gamma radiation is one of the rare examples of using a detector in the recombination region.
Compensated ionization chambers are most frequently used in nuclear reactors for their ability to measure a very wide range of neutron fluxes, spanning up to ten orders of magnitude. Such detectors are said to have a very large dynamic range.

**Fission chamber**
A fission chamber is an ionization chamber which has a window rimmed with a thin foil of $^{235}$U or $^{239}$Pu. After absorbing thermal neutrons, some nuclei of the fissile nuclide split; the resulting fission products are heavily ionized and have considerable kinetic energy, which is why the fission chamber generates a very strong and distinct signal.

### 3.6 Questions

1. Why are some types of radiation called directly ionizing and others indirectly ionizing?
2. How do we distinguish between the positive and negative charge produced by ionization in a gas detector?
3. How do we measure the quantity of charge released?
4. Which gas detectors allow us to deduce the energy deposited by radiation in the detector from the height of the voltage pulse?
5. What kind of measurements are scintillation detectors used for?
6. Describe the measurement procedure for a TL dosimeter.
7. What kind of information about neutrons does a BF$_3$ detector give?
8. Why are BF$_3$ detectors used in the proportional region of operation?
9. Why does a compensated ionization chamber have two parts?
4 DOSIMETRIC QUANTITIES

Learning objectives
After completing this chapter, the trainee will be able to:
1. Define the absorbed dose, absorbed dose rate and associated
   units.
2. Calculate the absorbed dose rate for cases of a point and line
   source.
3. Define radiation weighting factors and give their values for
   alpha, beta and gamma radiation and the range of values for
   neutrons.
4. Define the equivalent dose, equivalent dose rate and associated
   units.
5. Define tissue weighting factors.
6. Define the effective dose and its units.
7. Define the collective dose and its units.
8. Define exposure, exposure rate and associated units.

4.1 Absorbed dose ($D$)

In order to discuss the issues of radiation and radiation protection, we
need a suitable set of physical quantities and units to quantitatively
describe radiation fields and their effects.

In the context of nuclear physics, we have already met activity, a
physical quantity defined as the number of radioactive disintegrations
per unit time. Although the activity of a source is related to the
exposure to the radiation emitted by the source, the primary concern is
how much of this radiation actually reaches us. The situation can be
compared to a loudspeaker heard from a certain distance. If the
loudspeaker’s power is compared to source activity, we still need a
quantity which corresponds to the volume at the location of the
listener.

The basic dosimetric quantity is called the absorbed dose, $D$, defined
as the energy of radiation deposited (absorbed) in unit mass:

$$D = \frac{\Delta E}{\Delta m}$$

The unit for absorbed dose is the gray, in abbreviated form Gy, which
is defined as:

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

A target thus receives a dose of 1 Gy if 1 J of ionizing radiation is
deposited in 1 kg of matter. More frequently used and smaller units
are the mGy and µGy.
The old unit, which is often found especially in U.S. literature and regulations, is the *rad* and is a hundred times smaller than the gray:

\[
1 \text{ rad} = 0.01 \text{ Gy} \\
1 \text{ Gy} = 100 \text{ rad}
\]

The **absorbed dose rate**, \( \dot{D} \), tells us how quickly the absorbed dose to the target increases.

\[
\dot{D} = \frac{\Delta D}{\Delta t}
\]

The absorbed dose rate is measured in units of Gy/h, mGy/h etc.

**Example:**
A dosimeter is exposed to radiation. At 9:00, the absorbed dose is 3.7 mGy, and at 16:00, it is 28.2 mGy. What is the dose rate?

**Answer:**
\[
\Delta D = 28.2 \text{ mGy} - 3.7 \text{ mGy} = 24.5 \text{ mGy} \\
\Delta t = 16 \text{ h} - 9 \text{ h} = 7 \text{ h} \\
\dot{D} = \frac{24.5 \text{ mGy}}{7 \text{ h}} = 3.5 \text{ mGy/h}
\]

**Dose rate for point sources of radiation**

In case of a point source of radiation with activity \( A_c \), radiation spreads from it isotropically, i.e. evenly in all directions. Let us consider the simplest example, where a radionuclide emits one photon with each decay. At distance \( r \) from the source, the photons spread evenly over the surface of a sphere with radius \( r \), so that \( S = 4\pi r^2 \). At this distance, the photon flux thus equals:

\[
\Phi = \frac{A_c}{4\pi r^2}
\]

The dose rate is the product of the number of photons absorbed in the target per unit time, and the energy they pass to the target on absorption. The number of photons absorbed in the target is proportional to the number of photons reaching the target, i.e. the photon flux. This means the dose rate is proportional to the photon flux:

\[
\dot{D} \propto \Phi \propto \frac{A_c}{r^2}
\]

* A source of radiation may be considered as a point source when its distance from the target is more than 3 times its dimension.
The absorbed dose rate is directly proportional to the point source activity and inversely proportional to the square of the distance from the source. The proportionality constant is called the $\Gamma$ constant:

$$D = \Gamma \frac{Ac}{r^2}$$

The gamma constant can be computed individually for each radionuclide, but in practice we look it up in tables. It is normally given for air. Its values for some of the more important radionuclides are shown in Table 4.1. Since the gamma constant is directly dependent on the mass absorption coefficient (which is approximately 12% greater for tissue than for air), the dose rates in tissue exceed those in air by a factor of 1.12 as well.

Table 4.1: Gamma constants for certain nuclides.

<table>
<thead>
<tr>
<th>nuclide</th>
<th>half-life</th>
<th>$\Gamma$ [Gy/h · m²/Bq]</th>
</tr>
</thead>
<tbody>
<tr>
<td>$^{16}$N</td>
<td>7.1 s</td>
<td>$3.49 \cdot 10^{-13}$</td>
</tr>
<tr>
<td>$^{24}$Na</td>
<td>15 h</td>
<td>$4.59 \cdot 10^{-13}$</td>
</tr>
<tr>
<td>$^{51}$Cr</td>
<td>27.7 days</td>
<td>$5.45 \cdot 10^{-15}$</td>
</tr>
<tr>
<td>$^{54}$Mn</td>
<td>312 days</td>
<td>$1.1 \cdot 10^{-13}$</td>
</tr>
<tr>
<td>$^{59}$Fe</td>
<td>44.5 days</td>
<td>$1.57 \cdot 10^{-13}$</td>
</tr>
<tr>
<td>$^{58}$Co</td>
<td>70.8 days</td>
<td>$1.46 \cdot 10^{-13}$</td>
</tr>
<tr>
<td>$^{60}$Co</td>
<td>5.27 years</td>
<td>$3.25 \cdot 10^{-13}$</td>
</tr>
<tr>
<td>$^{85}$Kr</td>
<td>10.7 years</td>
<td>$4.74 \cdot 10^{-16}$</td>
</tr>
<tr>
<td>$^{95}$Zr</td>
<td>64 days</td>
<td>$1.1 \cdot 10^{-13}$</td>
</tr>
<tr>
<td>$^{131}$I</td>
<td>8.04 days</td>
<td>$6.71 \cdot 10^{-14}$</td>
</tr>
<tr>
<td>$^{134}$Cs</td>
<td>2.06 years</td>
<td>$2.37 \cdot 10^{-13}$</td>
</tr>
<tr>
<td>$^{137}$Cs</td>
<td>30 years</td>
<td>$7.56 \cdot 10^{-14}$</td>
</tr>
</tbody>
</table>

Dose rate for other source geometries

We also frequently deal with line sources of radiation, e.g. a pipe carrying radioactive fluid. The dose rate falls inversely proportionally to the distance from the line source:

$$\dot{D} = \pi \frac{\Gamma ac}{d}$$

where $ac$ is the specific activity (activity per unit of source length), and $d$ is the distance from the source.
4.2 Equivalent dose ($H$)

Things would be simple if radiation effects were directly proportional to the absorbed dose. Unfortunately, they are not. Besides the energy deposited in the body by radiation, the degree of biological damage also depends on the type of incoming radiation and sometimes on the energy of the particles.

Biological effects increase with the stopping power of particles which is also called linear energy transfer – LET. This means that particles with a large LET (alpha particles, neutrons) cause more damage than particles with a small LET (beta and gamma particles) at the same absorbed dose.

Since there is no easy way of adequately expressing the dependence of biological effects on linear energy transfer by a simple mathematical formula, we need to introduce the radiation weighting factor $w_R$ which accounts for the differences in the quality (effectiveness) of various radiations.

To estimate the degree of biological damage, we introduce a new dosimetric quantity, called the equivalent dose, $H$, defined as the product of the absorbed dose and the radiation weighting factor $w_R$:

$$H = w_R \cdot D$$

The unit for equivalent dose is called the sievert (Sv). This unit is used to point out that we are dealing with equivalent and not absorbed dose (the unit of the latter is Gy), although both can be expressed with the same basic physical units:

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

### Table 4.2: Radiation weighting factors (ICRP 2007).

<table>
<thead>
<tr>
<th>type of radiation</th>
<th>energy</th>
<th>radiation weighting factor $w_R$</th>
</tr>
</thead>
<tbody>
<tr>
<td>alpha ($\alpha$)</td>
<td>all energies</td>
<td>20</td>
</tr>
<tr>
<td>beta ($\beta$)</td>
<td>all energies</td>
<td>1</td>
</tr>
<tr>
<td>gamma ($\gamma$)</td>
<td>all energies</td>
<td>1</td>
</tr>
<tr>
<td>neutrons (n)</td>
<td>continuous function of energy</td>
<td>2.5 ~ 20</td>
</tr>
</tbody>
</table>

The radiation weighting factor, $w_R$, for beta and gamma radiation equals 1 and its value for other types of radiation is greater than 1 (Table 4.2).
An equivalent dose of 1 Sv is a very large dose, so we often use smaller units, mSv and µSv.

The old unit for equivalent dose, which is still encountered in American literature and legislation, is the rem (roentgen equivalent man):

\[ 1 \text{ rem} = 0.01 \text{ Sv} = 10 \text{ mSv} \]

\[ 1 \text{ Sv} = 100 \text{ rem} \]

**Example:**

An organ with a mass of 70 g has absorbed 385 GeV of alpha radiation. What is the equivalent dose received?

**Answer:**

\[
D = \frac{\Delta E}{\Delta m} = \frac{(385 \cdot 10^9 \cdot 1.6 \cdot 10^{-19} \text{ J})}{(70 \cdot 10^{-3} \text{ kg})} = 8.8 \cdot 10^{-7} \text{ Gy} = 0.88 \mu\text{Gy}
\]

\[
H = D \cdot w_r = (0.88 \cdot 20) \mu\text{Sv} = 17.6 \mu\text{Sv}
\]

The equivalent dose rate, \( \dot{H} \), tells us how quickly the equivalent dose to the target increases:

\[
\dot{H} = \frac{\Delta H}{\Delta t} = \dot{D} \cdot w_r
\]

The units used for equivalent dose rate are Sv/h, mSv/h and µSv/h.

**Example:**

What equivalent dose is received by a worker working for one year in a laboratory where the equivalent dose rate is 1.5 µSv/h? Assume that there are 2000 working hours in a year.

**Answer:**

\[
H = \dot{H} \cdot t = 1.5 \mu\text{Sv/h} \cdot 2000 \text{ h} = 3000 \mu\text{Sv} = 3 \text{ mSv}
\]

### 4.3 Effective dose (E)

Quite clearly the biological consequences vary if a given equivalent dose is received by the whole body or just by a part of the body or an organ. In terms of biological effects, we also find that not all organs are equally sensitive to radiation, i.e. the same equivalent dose produces more harmful effects in some organs than in others.

This is why another quantity was introduced, the effective dose, \( E \), which basically expresses the share of whole body effects contributed by a particular organ or tissue.

\[
E = w_T \cdot H ,
\]

where \( w_T \) is the **weighting factor for an organ or tissue** (T). If several organs are irradiated, each with its own dose, \( H_T \), the total effective dose equals:
\[ E = \sum_{T} w_T H_T \]

Table 4.3 indicates the values of the tissue weighting factor, \( w_T \). As for equivalent dose, the unit for effective dose is the Sievert (Sv). It should be emphasized that using the concept of effective dose only makes sense at low doses (\( E \ll \text{Sv} \)) and that tissue weighting factor values also apply to low doses only.

**Table 4.3:** Organ or tissue weighting factors (ICRP 2007).

<table>
<thead>
<tr>
<th>tissue or organ</th>
<th>tissue weighting factor ( w_T )</th>
</tr>
</thead>
<tbody>
<tr>
<td>breast</td>
<td>0.12</td>
</tr>
<tr>
<td>bone marrow</td>
<td>0.12</td>
</tr>
<tr>
<td>colon</td>
<td>0.12</td>
</tr>
<tr>
<td>lung</td>
<td>0.12</td>
</tr>
<tr>
<td>remainder</td>
<td>0.12</td>
</tr>
<tr>
<td>stomach</td>
<td>0.12</td>
</tr>
<tr>
<td>gonads</td>
<td>0.08</td>
</tr>
<tr>
<td>bladder</td>
<td>0.04</td>
</tr>
<tr>
<td>liver</td>
<td>0.04</td>
</tr>
<tr>
<td>oesophagus</td>
<td>0.04</td>
</tr>
<tr>
<td>thyroid</td>
<td>0.04</td>
</tr>
<tr>
<td>bone surfaces</td>
<td>0.01</td>
</tr>
<tr>
<td>brain</td>
<td>0.01</td>
</tr>
<tr>
<td>salivary glands</td>
<td>0.01</td>
</tr>
<tr>
<td>skin</td>
<td>0.01</td>
</tr>
<tr>
<td><strong>TOTAL</strong></td>
<td><strong>1.00</strong></td>
</tr>
</tbody>
</table>

### 4.4 Collective dose \( (E_c) \)

We also often want to indicate the overall effective dose received by a group of people. This quantity is called the collective dose, \( E_c \), and its unit is the *man·sievert* (abbreviated to man·Sv). Its calculation is quite straightforward as shown in the following example.

**Example:**

In a nuclear accident in a town with 2000 inhabitants, one half of the people received a dose of 20 mSv and the other half 10 mSv. What is the collective dose?

**Answer:**

\[
E_c = 1000 \text{ people} \cdot 20 \text{ mSv} + 1000 \text{ people} \cdot 10 \text{ mSv} = 30 \text{ man·Sv}
\]

If we generalize the above calculation, collective dose can be
expressed as:

\[ E_c = \sum_i n_i E_i , \]

where \( n_i \) is the number of people that received dose \( E_i \), or, dose \( E_1 \), was received by \( n_1 \) people, dose \( E_2 \) by \( n_2 \) etc.

### 4.5 Exposure (\( X \))

Historically the oldest radiological quantity is exposure but it is no longer used. It is defined by the ionization caused in air by X radiation or gamma radiation:

\[ X = \frac{Q}{\Delta m} , \]

where \( Q \) is the sum of the electric charges of all ions of the same sign released in a unit of mass, \( \Delta m \). The unit for exposure is As/kg air.

The relation to the old unit, the roentgen (R) is:

\[ 1 \text{ R} = 2.58 \cdot 10^{-4} \text{ As/kg} \]

Inversely:

\[ 1 \text{ As/kg} = 3876 \text{ R} \]

**Exposure rate**, \( \dot{X} \), tells us how quickly the charge of released electrons increases, in other words, what current flows from the air mass \( \Delta m \):

\[ \dot{X} = \frac{\Delta X}{\Delta t} = \frac{I}{\Delta m} \]

The unit for exposure rate is A/kg air, but it is more common to use the old units: R/s, R/h, mR/h etc.

### 4.6 Exercises

1. What is the absorbed dose in an organ with a volume of 40 cm\(^3\) and a density of 0.93 g/cm\(^3\) which absorbs \( 3 \cdot 10^5 \) MeV radiation?
2. How many alpha particles of 6 MeV energy cause an absorbed dose of 10 µGy in tissue with a mass of 70 g?
3. A point source of \(^{137}\text{Cs}\) has an activity of 1200 MBq. What is the absorbed dose rate in air 5 m away from the source?
4. Assess the activity of a \(^{60}\text{Co}\) source for which the dose rate measured at a distance of 3 m is 0.1 mSv/h.
5. Calculate the dose rate at 50 cm distance from a point source of:

---

*Exposure (\( X \)) and exposure rate (\( \dot{X} \)) are no longer official units*
a) 2 GBq $^{51}$Cr, b) 2 GBq $^{24}$Na.

6. Somebody left a 370 MBq source of $^{24}$Na in the laboratory by mistake. An employee unaware of this source spent 8 hours at her desk which is 2 m away from the source. Assess the dose she received.

7. The activity of a $^{60}$Co point source is 16 MBq. In how much time is the annual effective dose, 5 mSv, reached at 1 m distance from the source?

8. What is the equivalent dose of alpha particles if the absorbed dose is 3.2 µGy?

9. What is the absorbed dose in an organ with a volume of 40 cm$^3$ and a density of 0.93 g/cm$^3$ which absorbs $3 \times 10^5$ MeV energy of thermal neutrons?

10. A body part received 0.15 mGy radiation with a weighting factor of 5 and 0.22 mGy radiation with a weighting factor of 10. What is the total absorbed dose? What is the total equivalent dose?

11. After a scan with radioactive iodine, $^{131}$I, the equivalent dose to the thyroid was 61.5 mSv. What was the effective dose to the patient?
### 5 BIOLOGICAL EFFECTS OF RADIATION

**Learning objectives**

After completing this chapter, the trainee will be able to:

1. Name the types of cells and list the main cell parts.
2. Name the main substances that form a cell and their rough percentages.
3. Describe the cell division process.
4. Explain how radiation directly and indirectly affects the cell.
5. Describe the difference between the effects of different kinds of radiation in cells.
6. Describe the time development of radiation damage in cells and tissue.
7. Describe and distinguish between somatic and genetic effects.
8. Distinguish stochastic and deterministic effects.
9. List the deterministic effects of radiation.
10. Explain the concept of median lethal dose.
11. List the stochastic effects of radiation.
12. Clarify the risk of stochastic effects.

#### 5.1 Basic cell biology

The basic building blocks of living organisms are cells. For a better understanding of how ionizing radiation affects people, we will take a brief look at the structure of cells and some vital processes that take place in them.

The body of an adult person consists of around $4 \cdot 10^{13}$ cells. Cells differ from each other both in function and size. Most cells are relatively small (around 10 µm), whereas nerve cells can be up to a metre long. Cells fall roughly into two families: somatic cells and sex cells. Almost all cells in the body are somatic cells, and sex cells or gametes are only relevant to reproduction. Gametes pass hereditary information from one generation to the next.

The cell may be small in size but it is an extremely complex system. A typical somatic cell is shown in Figure 5.1. The outer boundary of a cell, called the cell membrane, both protects it from external factors and also forms a link between the cell and its environment. Most of the cell’s volume is filled with cytoplasm, a transparent mixture of water, various molecules and electrolytes. Also floating in cytoplasm are a number of structures called organelles, in which various metabolic and other processes take place. The most important organelle is the cell nucleus, which is the central element of the cell and controls all its vital functions.

Cells are composed of ~70-85 % water, ~10-20 % proteins, ~10 % carbohydrates (sugars) and ~2-3 % lipids (fats) and small quantities of
People do not live with the same cells from birth to death; human cells grow and regenerate. These processes are possible because cells divide. Cell division is called mitosis. Precise cell reproduction is ensured by the genetic material contained in the cell nucleus. During the resting phase (when the cell is not dividing), the genetic material is called chromatin. This is a complex of the intertwined strands of deoxyribonucleic acid (DNA) macromolecules and certain proteins. A DNA molecule has the characteristic shape of a double helix (similar to an “infinite” ladder), formed by two long entwined chains consisting of basic building blocks called nucleotides. The double helix strings together pairs of nucleotides (one on either side of the “rungs”) with one of the four organic bases: adenine (A) and thymine (T), or guanine (G) and cytosine (C). In the resting phase, chromatin controls the synthesis of proteins which give the cell its characteristic features (shape and function).

During cell division chromatins can be observed under microscope as chromosomes. They are spiralled, densely packed DNA. Human beings have exactly 46 chromosomes in each somatic cell and exactly 23 chromosomes in each sex cell. Before cell division chromatins (chromosomes) are doubled, and later on during mitosis distributed to daughter cells. Therefore each new cell acquires one copy of each chromosome. This ensures the transmission of genetic information about cell structure and functions to the next generation of cells.

The time between two cell divisions is called the cell cycle. Cell cycle length varies according to the type of cell. Cells of the human intestine divide approximately every 24 hours, while nerve and muscle cells
basically no different from the effects caused by various chemicals. Practically never.

5.2 Radiation effects on cells

No atom in a cell is free; all atoms are bound into molecules. Since molecular bonds are formed by electrons, ionization results in the breakdown of molecules or the formation of free radicals from previously neutral molecules.

When ionization causes a complex organic molecule in one of the organelles to split, this is a **direct effect of ionizing radiation** on the cell.

Since cells to a large extent (70-85 %) consist of water, it is more likely for radiation to ionize the water in the cell than to affect any of the (vital) organic molecules. Water ionization involves several chemical reactions called **water radiolysis**. Particularly important products of radiolysis include free hydroxyl radicals, OH⁻, and hydrogen peroxide, H₂O₂. Both are strong oxidants which are able to travel approximately 4 nm from their point of origin, where they have a very aggressive effect on cell elements. This process is called an **indirect effect of ionizing radiation** and its consequences are very similar to those of the direct radiation effect. In forms of radiation that rarely ionize matter (have a low LET), the indirect effect accounts for around 70 % of damage; for high-LET radiation, this share is smaller than the direct effects. It is important to note that both processes are essentially chemical. The biological effects of radiation are thus basically no different from the effects caused by various chemicals. Radiation is just one of the ways in which harmful chemical substances are formed in or enter the body (Figure 5.2).

![Figure 5.2: Direct and indirect radiation effects on cells.](image-url)
The cell is such a complex system that, even at the cellular level, there is no single mechanism through which radiation acts on the body. Radiation effects depend on which of the cell organelles is affected and what its role is in cell functioning. Although not all the details are understood as yet, many studies suggest that the most sensitive part of the cell is its nucleus, specifically the chromosomes. Of course, radiation may also damage other organelles. If damage is caused to a large molecular group, e.g. in the mitochondrion, the function of this mitochondrion will very likely be impaired or fail entirely. However, there are several mitochondria in most cells which can take over its role. On the other hand, damage to the cell DNA affects the function of the cell as a whole and may be transmitted to its progeny by mitosis. DNA is therefore a critical target for ionising radiation.

As it turns out, many cells damaged by radiation continue to function. During mitosis, however, a defective chromosome prevents them from dividing properly and they die. This is called mitotic death. Radiation effects are therefore most dramatic in organs with the most quickly dividing cells (red marrow, large intestine epithelium, skin epithelium, hair follicles). In contrast, tissues which are more static such as muscles and bones are less sensitive to radiation. Also very sensitive to radiation are embryos, especially in the first months after conception, because their cells divide at a much faster rate than in adults.

![Figure 5.3: Survival of cells exposed to acute doses.](image)

Parts of certain animal tissues can be removed and then grown in cell colonies which multiply spontaneously by cell division. Exposing such a colony to a short-term, acute dose of radiation enables very precise measurement of cell death as a function of dose. Figure 3.3 shows the proportion of surviving cells as a function of the absorbed dose. This figure very clearly indicates that radiation with high stopping power or a high linear energy transfer (LET), e.g. alpha radiation or neutron radiation, causes much greater biological damage.
than radiation with a low LET such as beta and gamma radiation at the same absorbed dose. This example illustrates why absorbed dose is insufficient to describe biological effects and why we also need to use the equivalent dose, which multiplies the absorbed dose by the relevant radiation weighting factor. Figure 5.3 also shows that the proportion of cells surviving irradiation with a high LET radiation falls exponentially (the chart uses a logarithmic scale for the vertical axis) with increasing dose, whereas in low-LET radiation, the proportion of surviving cells decreases very slowly at lower doses and only starts falling exponentially at high doses. Although the process of cell death is not yet fully understood, this suggests that some kind of repair mechanism exists in cells at low doses and low-LET radiation which reduces the number of cells killed.

In certain cases, the damaged cells keep the ability to divide further. As they divide, the change in their chromosomes is passed to the next generations of cells. This phenomenon is called mutation. Just as the mechanism of the effects of ionizing radiation on the cell is basically no different from the mechanism that underlies the effects of other environmental factors, mutations in the body are by no means caused by radiation alone; they are so to speak a daily occurrence produced by various external influences on cells.

Alternatively, mutations may cause pathological changes which are manifested as the abnormal functioning of individual organs or as a greatly increased rate of cell division in a tissue or organ. Such a pathological change is called cancer.

In a way, it is paradoxical that radiation may cause cancer, while on the other hand, radiation is also one of the most powerful tools for treating cancer. The rapid division of cancerous cells makes them much more sensitive to radiation than healthy tissue. Strong irradiation of the cancerous growth destroys the cancerous cells, whereas the cells in the surrounding healthy tissue largely survive. Of course, every effort is made to confine the beam of radiation to the diseased tissue.

If any chromosome in the sex cells (in the sex glands – gonads) is damaged, the mutation is transmitted to the offspring and it is only in them – and/or their offspring – that certain pathological changes will develop. This process is called genetic change.

Radiation damage to organisms consists of several phases of varied duration:

- **physical phase**: \ (~10^{-12} \text{s}: ionization
- **chemical phase**: \ (~10^{-6} \text{s}: formation and diffusion of free radicals
  \ (~10^{-3} \text{s}: DNA damage
- **biological phase**: \ (~seconds: DNA repair
  \ (~hours: cell death
  \ (~days: mutations, cell transformation,
  \ chromosome aberrations
  \ (~years: development of cancer, hereditary
5.3 Radiation effects on humans

The effects of radiation on humans can be classified by two criteria. The first classification refers to the type of cells damaged or to the person that suffers the effects of radiation. Symptoms which are a consequence of damaged somatic cells and occur in the irradiated person are called **somatic effects**. Symptoms which are a consequence of damaged gonads and occur in the irradiated person’s offspring are called **genetic effects**.

The second classification relates to the probability that the effect or medical condition will occur. Effects as a consequence of radiation which are certain to occur above a specific dose are called **deterministic effects**. These effects are a consequence of irreparable damage to cells and their death during unsuccessful division. Effects which occur with a certain probability only in a specific percentage of irradiated persons are called **stochastic effects**. These effects are a consequence of accumulated mutations in irradiated cells.

It should be emphasized that the effects of radiation depend not only on the equivalent or effective dose received, but also on how this dose was received:
- Acute: a high dose in a short time interval.
- Chronic: prolonged irradiation with low doses.
- Local: exposure of a particular organ.
- Total: whole body irradiation.

Most information about the effects of radiation is drawn from cases of acute irradiation with high doses (e.g. survivors of the Hiroshima and Nagasaki bombings or accidents involving radiation sources). However, what is most relevant for the practical application of radiation protection principles, especially to occupationally exposed workers, is multiple (chronic) low dose irradiation. Since we know that repair mechanisms are at work at low doses, we may assume that an assessment of harmful radiation effects based on data from acute doses is a conservative assessment of the effects of chronic exposure to the same dose.

**Deterministic effects of radiation**

As the name tells us, deterministic effects are effects which are certain to occur following irradiation with a high enough dose. They will manifest after a short delay, hours to weeks after irradiation. They are also characterized by a **dose threshold**, i.e. the minimum dose, $D_0$, at which these effects occur. Below this threshold there are no radiation effects, and above it their severity or intensity increases with the dose received (Figure 5.4). Deterministic effects can also be unambiguously linked to excessive exposure to radiation.
Deterministic effects include:
- erythema (skin redness),
- epilation (hair loss),
- reduced number of blood cells (resulting from bone marrow damage),
- cataract,
- nausea, vomiting, diarrhoea,
- damage to children (deformation of organs, mental retardation),
- temporary or permanent sterility.

The severity of deterministic effects as a function of received dose is given in Table 5.1.

**Table 5.1:** Deterministic effects of acute whole-body irradiation.

<table>
<thead>
<tr>
<th>Dose (Gy)</th>
<th>Probable effect</th>
</tr>
</thead>
<tbody>
<tr>
<td>0 – 0.5</td>
<td>No prominent effects, possible minor changes in blood count.</td>
</tr>
<tr>
<td>0.6 – 1.2</td>
<td>Nausea and vomiting occur after the first day in 5-10 % of the cases. No serious incidence of disease.</td>
</tr>
<tr>
<td>1.3 – 1.7</td>
<td>The first day is followed by nausea and vomiting. 25 % cases display symptoms of radiation sickness. No fatal cases.</td>
</tr>
<tr>
<td>1.8 – 2.2</td>
<td>In 50 % of the cases, nausea and vomiting occurs within the first day. The symptoms of radiation sickness are more prominent. No fatal cases are expected.</td>
</tr>
<tr>
<td>2.3 – 3.3</td>
<td>In 100 % of the cases, nausea and vomiting occurs within the first day. Serious signs of radiation sickness appear.</td>
</tr>
<tr>
<td>3.4 – 5.0</td>
<td>Median lethal dose, LD$_{50/30}$ – 50 % deaths in 30 days.</td>
</tr>
<tr>
<td>&gt; 100</td>
<td>Radiation syndromes. 100 % deaths with a time delay</td>
</tr>
</tbody>
</table>

**Figure 5.4:** Deterministic effects of radiation.
Protection against deterministic effects is relatively simple – by ensuring that the doses to individual organs are lower than the threshold dose, the risk of any pathological changes developing is prevented.

**Stochastic effects of radiation**

Stochastic effects of radiation are effects which can only be predicted to occur with a certain probability. In this case, a threshold dose, i.e. a minimum dose below which there is no risk of these effects, cannot be proved. A characteristic feature of stochastic effects is that the probability of a medical condition occurring rises with increasing dose, whereas the severity or intensity of an effect (in the case when it occurs) is not related to dose. Furthermore, the medical condition cannot be unequivocally linked to radiation exposure; all that may be determined is a statistical increase in the medical condition beyond its natural frequency or incidence due to other causes. Stochastic effects of radiation include in particular:

- cancer,
- genetic effects.

**Cancer** is (after cardiovascular diseases) the second leading cause of death. In the developed world, around 40% of people contract cancer and almost 20% die from cancer. Cancer is not a single disease but a group of around a hundred very similar diseases, all of which share rapid, uncontrolled division of cells. Over 90% of cancers are caused by various chemicals which are called **carcinogens**. More than a thousand carcinogens are known and most of them occur naturally in the environment. Radiation is therefore just one – albeit not chemical – of many carcinogens. Among the 86572 surviving victims of the Hiroshima and Nagasaki bombings, 249 people died of leukaemia and 7578 of other types of cancer, but a comparison with other groups of people shows that only 87 of the leukaemia cases and 334 of the cases of other cancer types may be attributed to radiation exposure, while the rest resulted from other, natural causes. This suggests that radiation is not a conspicuously strong carcinogen.

The bulk of the data on additional incidence of cancer was obtained by monitoring Hiroshima and Nagasaki survivors and from some accidents involving sources of radiation. What is characteristic of all these cases is that the individuals received relatively high radiation doses. It was also found that, at high doses, the probability of developing cancer is proportional to the dose received. For low doses, the number of additional disease cases was too small to be statistically identified. It was therefore assumed that even in low doses, the risk of

cancer is proportional to the dose received and that there is no “safe dose” or threshold dose with no risk below that value. However, there are many indications that certain repair mechanisms exist at low doses which reduce this risk. Taking a conservative approach, however, radiation protection regulations are based on the linear no-threshold hypothesis (dotted section of the curve in Figure 5.5).

![Stochastic effects of radiation](image)

**Figure 5.5:** Stochastic effects of radiation.

Leukaemia is the leading cancerous disease caused by radiation, followed by breast, lung, thyroid and stomach cancer. For each of these, the probability of occurrence can be determined and expressed as the number of expected deaths per specific collective dose. The total probability for any type of cancer is 5 % per 1 man·Sv.

**Example:**
Calculate the number of additional deaths from cancer in a population of 100,000 people exposed to an average dose of 10 mSv.

**Answer:** The collective dose is:

\[ E_c = 100,000 \text{ people} \cdot 0.01 \text{ Sv} = 1000 \text{ man·Sv} \]

The expected additional number of cancer cases due to radiation is 5 % of the collective dose, therefore 50 people. This number should be set against the 20,000 people in this population who will die of cancer from other causes.

What is also characteristic of all types of cancer is a latent period, i.e. the time between radiation exposure and the onset of cancer. In leukaemia the latent period is 5-15 years, and for other types of cancer it is 10 years or more.

**Genetic effects** are another type of stochastic effect of radiation. These are effects which are only displayed in the offspring of the irradiated person. Being even less probable than cancer, there has been so far no statistically proven increase of genetic effects in any group of irradiated persons.
5.4 Questions

1. What is the difference between the direct and indirect effects of ionizing radiation in tissue cells?
2. What are the basic characteristics of deterministic and stochastic effects of ionizing radiations?
3. What are the symptoms of radiation sickness and at roughly what doses do changes occur in the body and on the skin?
6 EXTERNAL RADIATION EXPOSURE

Learning objectives
After completing this chapter, the trainee will be able to:
1. Describe the difference between external and internal exposure.
2. Give and describe examples of potential external exposure.
3. Describe methods of protection against external exposure.
4. Describe the use of time as a means of protection.
5. Describe the effect of distance on dose rate.
6. Describe how the type of source is taken into account.
7. Describe the influence of shielding on gamma radiation.
8. Describe the build-up of gamma radiation.
9. Calculate the effect of distance, time and shielding on gamma radiation dose rate.
10. Describe shielding for sources of beta rays.
11. Explain the basic characteristics of neutron shields.

6.1 Types of radiation exposure

The main objective of radiation protection, as its name implies, is protection against excessive exposure to radiation. This involves the important step of calculating or assessing the received dose or dose rate which depends on the configuration of the radiation source. The methods of protection as well as the calculation of the dose received very much depend on whether the source is located inside or outside the human body and on the type of radiation emitted by the source.

When the source of radiation is located outside the body, this produces external exposure. In this case, alpha radiation is harmless, because it is already stopped by the dead outer layer of the skin. Similarly, beta radiation is only harmful in some special cases (direct exposure to unshielded sources, skin contamination, direct eye irradiation) and safety is achieved just by wearing clothes and other protective items (safety spectacles, gloves). Thus, when dealing with external sources of radiation, it is most important to provide protection against gamma and neutron radiation. Since – except in some special cases – a reactor is a source of neutron radiation only during operation, the most important consideration in practice is protection against gamma radiation.

6.2 Time, distance, shielding

Exposure to radiation may in principle be reduced by three parameters:
- time,
- distance,
- shielding.

**Time**

The less time is spent in a field of radiation, the smaller is the dose received. This time can be shortened, particularly with proper preparation and practice outside the radiation area. This is to ensure that the actual exposure time is the shortest time required to perform the job and that the chances of a mistake (and the need to do the same job again) are minimal. Examples of good practice for reducing time are:

- precise planning and organisation of the work (breaking down the tasks beforehand),
- meetings before the work is started (to plan and coordinate activities),
- training on a mock-up,
- preparing tools in a clean environment,
- optimizing protective clothing,
- video surveillance,
- remote communication,
- delaying the beginning of work (decay of short-lived nuclides),
- ensuring appropriate working conditions (ventilation, lighting, appropriate supply of air).

**Distance**

The radiation dose rate falls with increasing distance: for point sources with the square of the distance, and for line sources proportionally with distance. The distance from the source of radiation may be increased by:

- using telescopic measuring instruments (teletectors),
- remote reading of instruments,
- using tongs, pincers, tweezers and other adapted tools,
- remote control of tools,
- using video surveillance,
- repairing portable parts in a hot machine shop,
- using remote communication,
- indicating hot spots and radiation levels,
- removing sources of radiation.

**Shielding**

Although distance and time can be put to good use as protection against external radiation, shielding is a more reliable method of reducing dose rate. In principle shielding alone could be used, but in practice the scope of shielding depends on the possibilities (of the location or equipment involved in a job), the actual effect on collective dose (setting up a shield also results in radiation exposure), as well as the ratio between its cost and the expected benefits (dose reduction). Shielding consists of:

- portable lead blankets or screens,
- lead bricks,
- water (primary system, secondary side of steam generators),
- existing concrete walls, etc.

Due to gamma ray scattering and bremsstrahlung, shielding calculations are not simple and an expert should be consulted on the topic.

**Example 1:**
It is presumed that a worker repairing a valve will be irradiated with gamma radiation at a rate of 1.5 mGy/h. How long can he work on the valve without receiving an effective dose higher than 0.5 mSv?

*Answer:*
We are dealing with gamma radiation, so the effective dose is identical to the absorbed dose. The time in which a particular dose is reached equals:

\[
t = \frac{D}{\dot{D}} = \frac{0.5 \text{ mGy}}{1.5 \text{ mGy h}^{-1}} = 0.33 \text{ h} = 20 \text{ min}
\]

**Example 2:**
One metre away from a point source of \(^{60}\text{Co}\) the dose rate is 1 mSv/h. What is it at a distance of 10 cm?

*Answer:*
If the distance is 10 times smaller, the dose rate is 100 times higher, thus 100 mSv/h. The problem is mathematically solved as follows:

\[
\dot{D} \cdot r^2 = \dot{D}_0 \cdot r_0^2 \Rightarrow \dot{D} = \dot{D}_0 \cdot \left(\frac{r_0}{r}\right)^2 = 1 \text{ mSv/h} \cdot (10)^2 = 100 \text{ mSv/h}
\]

The next sections describe how appropriate shields are determined for gamma or beta radiation and for neutrons.

### 6.3 Shielding from gamma radiation

When discussing the interaction of gamma radiation with matter we learned that the intensity \(j\) of monoenergetic gamma radiation falls exponentially with shielding thickness:

\[
j = j_0 \cdot 2^{-\frac{d}{d_{1/2}}},
\]

where \(d_{1/2}\) is the **half-value layer** of the shielding material. The half-value layer represents the thickness of a material for which the intensity of monoenergetic gamma radiation is reduced by a factor of 2. Similarly, the **tenth-value layer**, \(d_{1/10}\) is the shielding thickness which attenuates radiation by a factor of 10.
Table 6.1: Conservative values of tenth-value layers for various substances.

<table>
<thead>
<tr>
<th>substance</th>
<th>$d_{1/10}$ [cm]</th>
</tr>
</thead>
<tbody>
<tr>
<td>water</td>
<td>60</td>
</tr>
<tr>
<td>concrete</td>
<td>25</td>
</tr>
<tr>
<td>steel</td>
<td>10</td>
</tr>
<tr>
<td>lead</td>
<td>5.6</td>
</tr>
</tbody>
</table>

Since in addition to directly transmitted radiation Compton-scattered photons also penetrate the shield, a slightly higher value of the tenth-value layer is used to give a conservative assessment of shielding effectiveness regardless of photon energy. Its values for radiation from spent fuel elements are shown in Table 6.1.

If the shielding thickness equals the one tenth-value layer, radiation is weakened by a factor of 10, at two such thicknesses by a factor of $10 \times 10 = 100$, at three thicknesses by 1000, etc.

**Example:**

On the surface of the spent fuel pit, the dose rate is 1 µSv/h. The fuel elements are immersed 6 m below the water level. What would the dose rate be at the same point if there was no water in the pit?

*Answer:* A 6-metre layer of water amounts to 10 tenth-value layers. Without a water shield the dose rate would thus be greater by a factor of $10^{10}$:

$$\hat{H} = \hat{H}_0 \cdot 10^{10} = 10^6 \text{ Sv/h} \cdot 10^{10} = 10^4 \text{ Sv/h}$$

The examples given provide merely a rough assessment of shielding effectiveness. For precise assessment or more complicated geometries, calculation is best left to the experts.

### 6.4 Shielding from beta radiation

Beta emitters contribute to potential external exposure in two ways: by the beta rays themselves, i.e. fast electrons (or positrons), and by the bremsstrahlung produced by the stopping of electrons in matter. Beta particles can be relatively easily stopped by using a shield with a thickness larger than their range.

In the case of protected beta emitters, bremsstrahlung often poses the only risk of external exposure. The share of bremsstrahlung is reduced by using a substance with a low atomic number, $Z$, for the beta radiation shield. In addition, the inner shield (designed to stop beta particles) can be covered with an outer shield made of a material with a high $Z$ which additionally absorbs bremsstrahlung.
6.5 Shielding from neutron radiation

Like gamma radiation, neutron radiation is highly penetrating. However, for several reasons, neutron shielding is rather more complicated than photon shielding. While cross-sections for photons vary evenly with energy and the atomic number of the substance, cross-sections for neutrons vary quite unsystematically from element to element and include complicated resonances depending on the energy of the neutrons. Moreover, in a great many substances, the values of cross-sections are better known for photons than for neutrons.

Since neutrons only react with matter through nuclear reactions, the effectiveness of a neutron shield is related to its neutron cross-section. However, a substance with a large cross-section is not enough to ensure good neutron shielding. Though scattering and absorption are included in the total cross-section, neutrons only disappear by absorption. On the other hand, nuclei normally have very small fast neutron absorption cross-sections, which is why neutron shields work on the principle of first slowing down the neutrons and then absorbing them with a substance which has a large thermal neutron capture cross-section. For radiative neutron capture, i.e. an \((n, \gamma)\) reaction, which releases high-energy gamma rays, the neutron shield must also contain a substance which absorbs photons. In this respect, a more favourable choice are neutron absorbers where the capture involves an \((n, \alpha)\) reaction, e.g. boron.

Neutron shields consist of the following components:
- a material which slows down fast neutrons (by elastic scattering on light nuclei), e.g. hydrogen in water, paraffin or concrete,
- a material which absorbs slow neutrons, e.g. boron, cadmium,
- a material which absorbs secondary \(\gamma\) rays, i.e. heavy elements, e.g. steel or special concretes containing heavy elements.

6.6 Exercises

1. A dose rate of 1 mSv/h is measured at 1 m distance from a source of \(\gamma\) radiation. At what distance from the source of radiation is its contribution equal to the natural background, 0.1 \(\mu\text{Sv/h}\)?
2. At a certain point, the dose rate is 200 mSv/h. What is the dose rate after placing 15 cm thick shielding between the source of radiation and the given point? The tenth-value layer of the shielding material is 5 cm.
3. A source of \(^{60}\text{Co}\) has an activity of \(3.7 \cdot 10^{10}\) Bq. What is the dose rate at a distance of 2 m? What is the dose rate at the same distance if a wall of 3 tenth-value layers is in between?
4. A pipe carrying a concentrate from an evaporator runs along the wall of a long room. Six metres from the pipe, the dose rate is
50 μGy/h. What is the dose rate at a distance of 1 m from the pipe?
7 INTERNAL RADIATION EXPOSURE

Learning objectives
After completing this chapter, the trainee will be able to:
1. Explain what behaviour may lead to internal exposure.
2. Define the effective half-life for internal exposure.
3. Explain dose coefficients and derived concentrations.
4. Calculate the committed effective dose for an intake of radioactive material into the body.
5. Describe exposure to radioactive noble gases.
6. Explain the methods of protection against internal exposure.
7. List basic protective clothes and their characteristics.
8. Explain the purpose of respiratory equipment.
9. Define the protection factor of respirators.
10. Calculate the required protection factor of respiratory equipment for a known content of radionuclides in air.
11. Explain what a “fit test” means.

7.1 The pathways of radionuclides into the body

Internal exposure or internal contamination is caused by the presence of radioactivity (radionuclides) in the body. Radionuclides can enter the body in 4 ways:
- with food and drink (ingestion),
- by breathing (inhalation),
- through the skin,
- through open wounds. \( \text{(absorption)} \)

In each of these cases, particular radionuclides follow separate pathways, so the dose received may also depend on the mode of intake. In contrast to external exposure, the greatest danger in internal exposure is posed by alpha emitters and certain beta emitters.

The concentration of radionuclides in the body diminishes over time - if there is no additional intake – due to their radioactive decay and biological elimination from the body through the gastrointestinal tract, urinary system and exhalation. Since the possibilities for artificially increasing the elimination of radionuclides from the body are very limited, the main objective of protection against internal exposure is to prevent any internal exposure from taking place at all.

Preventing internal exposure starts by preventing contamination of the environment in which people live and work. However, if a (working) area is already contaminated, any work in the area is subject to special measures. In the case of a particularly highly contaminated environment or a high risk of internal contamination, personal protective equipment (face masks, protective clothing) is used.
In contrast to external exposure, the dose received from internal exposure cannot be directly measured and can only be calculated or estimated.

### 7.2 Effective half-life, committed effective dose

Different organs vary in their susceptibility to different chemical elements. It is known that e.g. iodine builds up in the thyroid gland, calcium in the bones, and so on. Some other elements disperse evenly throughout the body. Once in the body, the radioactive isotopes of individual elements accumulate in respective target organs. Alpha or beta radiation emitted by the respective radionuclides stops in the target organ and causes only a dose to the latter, whereas gamma radiation also irradiates other parts of the body from a particular organ and also escapes from the body.

No substance, whether radioactive or not, stays in the body indefinitely. Ordinary biological processes remove all substances from the organs; their concentration falls roughly exponentially:

\[
\alpha \approx 2^{-\frac{t}{t_{1/2b}}},
\]

where \(t_{1/2b}\) is the biological half-life. Its value is determined experimentally and for some radionuclides it also depends on the target organs (Table 7.1).

#### Table 7.1: Biological half-lives.

<table>
<thead>
<tr>
<th>radionuclide</th>
<th>organ</th>
<th>(t_{1/2b})</th>
</tr>
</thead>
<tbody>
<tr>
<td>(^{3}\text{H}) (in water)</td>
<td>whole body</td>
<td>12 days</td>
</tr>
<tr>
<td>(^{60}\text{Co})</td>
<td>whole body</td>
<td>10 days</td>
</tr>
<tr>
<td>(^{90}\text{Sr})</td>
<td>whole body</td>
<td>36 years</td>
</tr>
<tr>
<td></td>
<td>bones</td>
<td>49 years</td>
</tr>
<tr>
<td>(^{131}\text{I})</td>
<td>whole body</td>
<td>138 days</td>
</tr>
<tr>
<td></td>
<td>thyroid</td>
<td>138 days</td>
</tr>
<tr>
<td>(^{137}\text{Cs})</td>
<td>whole body</td>
<td>70 days</td>
</tr>
<tr>
<td></td>
<td>bones</td>
<td>140 days</td>
</tr>
</tbody>
</table>

If the substance in the body is radioactive, its activity declines as a result of both biological processes and radioactive decay:

\[
Ac = Ac_0 \cdot e^{-\frac{t}{t_{1/2}}} \cdot e^{-\frac{t}{t_{1/2b}}} = Ac_0 \cdot e^{-\left(\frac{1}{t_{1/2}} + \frac{1}{t_{1/2b}}\right)} = Ac_0 \cdot e^{-\frac{t}{t_{1/2b}}},
\]

where \(t_{1/2}\) is the “usual” radioactive half-life. The decline in the
activity of a radionuclide in the body is therefore defined by the **effective half-life**, which is calculated by the following formulas:

\[
\frac{1}{t_{1/2E}} = \frac{1}{t_{1/2}} + \frac{1}{t_{1/2b}}
\]

\[
t_{1/2E} = \frac{t_{1/2} \cdot t_{1/2b}}{t_{1/2} + t_{1/2b}}
\]

The effective half-life is therefore the time in which the activity of a radionuclide in the body diminishes to half its initial value.

**Example:**
Calculate the effective half-life of \(^{90}\text{Sr}\) in bone. Its radioactive half-life is 28.6 years.

**Answer:**

\[
t_{1/2E} = \frac{t_{1/2} \cdot t_{1/2b}}{t_{1/2} + t_{1/2b}} = \frac{28.6 \text{ years} \cdot 49 \text{ years}}{28.6 \text{ years} + 49 \text{ years}} = 18 \text{ years}
\]

Calculating the equivalent dose rate caused by a radionuclide in the body is very complicated. It depends on its activity, mode of intake, the type and energy of radiation, the chemical form of the substance containing the radionuclide and the weighting factors of target organs.

However, the dose rate is rarely immediately relevant when dealing with internal contamination. As a rule, radionuclides cannot be artificially removed from the body, so the body remains exposed to internal radiation until their decay and biological elimination – in practice, this is approximately seven effective half-lives. When this period exceeds life expectancy (this applies e.g. to strontium-90), the irradiation period is held to be 50 years for adults and 70 years for children. The dose received in this period is called the **committed effective dose**. From the perspective of radiation protection, it represents the most important dosimetric quantity of internal exposure. It is always calculated and added to annual effective dose in the year of intake of the radionuclide.

### 7.3 Dose coefficient, derived air concentration

Committed effective doses have been calculated for all radionuclides, individually for breathing (inhalation) and swallowing (ingestion). The quotient that links the committed effective dose and the intake of activity is called the **dose coefficient**, \(h(g)\). The dose coefficient values for some radionuclides are given in Table 7.2.
Table 7.2: Dose coefficients \( h(g) \) for inhalation and ingestion.

<table>
<thead>
<tr>
<th>Radionuclide</th>
<th>Half-life</th>
<th>( h(g) ) [Sv/Bq]</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td></td>
<td>inhalation</td>
</tr>
<tr>
<td>(^{3})H</td>
<td>12.3 years</td>
<td>( 1.8 \times 10^{-11} )</td>
</tr>
<tr>
<td>(^{51})Cr</td>
<td>27.7 days</td>
<td>( 3.6 \times 10^{-11} )</td>
</tr>
<tr>
<td>(^{54})Mn</td>
<td>312 days</td>
<td>( 1.5 \times 10^{-9} )</td>
</tr>
<tr>
<td>(^{59})Fe</td>
<td>44.5 days</td>
<td>( 3.5 \times 10^{-9} )</td>
</tr>
<tr>
<td>(^{58})Co</td>
<td>70.8 days</td>
<td>( 2.0 \times 10^{-9} )</td>
</tr>
<tr>
<td>(^{60})Co</td>
<td>5.27 years</td>
<td>( 2.9 \times 10^{-8} )</td>
</tr>
<tr>
<td>(^{90})Sr</td>
<td>29.1 years</td>
<td>( 1.5 \times 10^{-7} )</td>
</tr>
<tr>
<td>(^{131})I</td>
<td>8.04 days</td>
<td>( 7.6 \times 10^{-9} )</td>
</tr>
<tr>
<td>(^{137})Cs</td>
<td>30 years</td>
<td>( 4.8 \times 10^{-9} )</td>
</tr>
<tr>
<td>(^{239})Pu</td>
<td>24100 years</td>
<td>( 4.7 \times 10^{-5} )</td>
</tr>
</tbody>
</table>

Legislation lays down the dose limits, \( E_m \), which may be received by workers or members of the public within a calendar year. Based on these dose limits we can use dose coefficients to determine the annual limit on intake (ALI) for a particular radionuclide:

\[
\text{ALI} = \frac{E_m}{h(g)}
\]

The derived air concentration (DAC) is defined as the concentration of a given radionuclide in air (or in water or food – in this case it is called the derived concentration) at which the annual limit on intake for the radionuclide would be reached by inhalation (or drinking water or by dietary intake).

When working with sources of radiation, the greatest challenge is to prevent the intake of radionuclides into the body by inhalation, which means that in practice, derived concentrations are of key importance. They are calculated on the basis of the following premises:

- a worker works 2000 hours a year,
- the breathing rate is 1.2 m\(^3\) of air per hour.

The derived air concentration is thus:

\[
\text{DAC} = \frac{\text{ALI}}{V_a},
\]

where \( V_a \) is the volume of air breathed by a worker during working time and so \( V_a = 1.2 \text{ m}^3/\text{h} \times 2000 \text{ h} = 2400 \text{ m}^3 \).
**Example:**
Calculate the DAC for \(^{131}\)I in air if the dose limit is 20 mSv.

*Answer:*  
\[
DAC = \frac{ALI}{V_s} = \frac{E_{\text{al}}}{h(g) \cdot V_s} = \frac{20 \cdot 10^{-3} \text{Sv}}{7.6 \cdot 10^{-3} \text{Sv Bq}^{-1} \cdot 10^6 \text{ m}^3} = 1100 \text{ Bq/m}^3
\]

The values of derived air concentrations for major radionuclides are given in Table 7.3.

**Table 7.3:** Derived air concentrations for workplaces.

<table>
<thead>
<tr>
<th>radionuclide</th>
<th>DAC [Bq/m(^3)]</th>
</tr>
</thead>
<tbody>
<tr>
<td>(^3)H (HTO)</td>
<td>460 000</td>
</tr>
<tr>
<td>(^{60})Co</td>
<td>290</td>
</tr>
<tr>
<td>(^{90})Sr</td>
<td>56</td>
</tr>
<tr>
<td>(^{131})I</td>
<td>1100</td>
</tr>
<tr>
<td>(^{137})Cs</td>
<td>1700</td>
</tr>
<tr>
<td>(^{239})Pu</td>
<td>0.18</td>
</tr>
</tbody>
</table>

### 7.4 Exposure to radioactive noble gases

Radionuclides which are the radioisotopes of noble gases do not form chemical bonds with substances in the body, so they are not taken in by breathing or food, at least not in a significant degree. If present in air, they enter the lungs on inhalation and leave them on exhalation. Any dose is chiefly due to external radiation from noble gases in the ambient air. Since the received dose is independent of biological processes, it is relatively easy to calculate. It is usually given as the effective dose received in one day due to the presence of radioactive noble gases in the air. Its values are shown in Table 7.4.

**Table 7.4:** Effective doses from exposure to noble gases.

<table>
<thead>
<tr>
<th>Noble gas radionuclide</th>
<th>(t_{1/2})</th>
<th>Effective dose received in one day per unit of concentration in air [Sv d(^{-1})/Bq m(^{-3})]</th>
</tr>
</thead>
<tbody>
<tr>
<td>(^{41})Ar</td>
<td>1.83 h</td>
<td>(5.3 \cdot 10^{-9})</td>
</tr>
<tr>
<td>(^{85})Kr</td>
<td>10.7 y</td>
<td>(2.2 \cdot 10^{-11})</td>
</tr>
<tr>
<td>(^{85m})Kr</td>
<td>4.48 h</td>
<td>(5.9 \cdot 10^{-10})</td>
</tr>
<tr>
<td>(^{133})Xe</td>
<td>5.24 d</td>
<td>(1.2 \cdot 10^{-10})</td>
</tr>
<tr>
<td>(^{135})Xe</td>
<td>9.1 h</td>
<td>(9.6 \cdot 10^{-10})</td>
</tr>
</tbody>
</table>
7.5 Protection against internal contamination

Protection against internal exposure is based on the principles of defence in depth:

- **Restriction of open sources of radiation**: Sources of radiation and processes which may cause radioactive contamination of surfaces or air should be restricted to a reasonably small number of designated areas.
- **Control of areas containing sources of radiation**: Areas where work takes place should be suitably architecturally designed by selecting surface materials that facilitate effective decontamination and ensuring controlled outlets and good ventilation with air filtration. Where necessary (the requirements are laid down by legislation), work is carried out within a controlled area. Access to this area is physically controlled, only workers meeting certain requirements (personal protective equipment, training, etc.) are allowed to work in this area and appropriate equipment for personal contamination monitoring and decontamination should be available at the exit.
- **Preventing internal contamination by ingestion and absorption**: While working in radiation controlled areas it is prohibited to eat, drink, smoke and work with open cuts or wounds.
- **Worker protection**: Workers are required to use personal protective equipment. Its purpose is to prevent skin contamination (*protective clothing*) and – in cases of increased air contamination – to restrict the intake of radioactive material by inhalation (*respiratory protective equipment*).

7.6 Protective clothing and equipment

The purpose of protective clothing is to intercept the contamination which would otherwise affect the worker’s skin or “normal” clothing, which is why protective clothing is always presumed to be contaminated. It is put on when entering the controlled area and taken off on leaving it. The clothing must be designed so that it can be taken off easily and without contaminating the skin or the clothes worn underneath. Workers should therefore be first trained to properly put on and take off their protective clothing.

In nuclear power plants, workers are also exposed to other industrial hazards: heat, noise, possibility of falling objects, chemicals and so on, which likewise call for special protective items.

Within the controlled area of a nuclear power plant, we always use **basic protective equipment** which (depending on the facility) consists of the following devices and clothes (Figure 7.1):

- a hard hat,
- underwear,
- cotton overalls,
- cotton gloves,
- protective shoes,
- a set of personal dosimeters (a daily – electronic and a monthly – TLD/OSL dosimeter).

**Figure 7.1:** A worker wearing basic protective equipment (example).

**Figure 7.2:** A worker in Tyvek coveralls with a respirator.
In areas where increased surface or air contamination is probable or confirmed, one or several pieces of **additional protective equipment** are also used according to the level of contamination:
- additional rubber gloves,
- shoe covers,
- Tyvek coveralls,
- special protective clothing,
- a respirator,
- a half mask,
- a mask,
- an air-supplied mask.

**Table 7.5:** Protective clothing in a controlled area (example).

<table>
<thead>
<tr>
<th>Type of clothing</th>
<th>washable</th>
<th>single use</th>
</tr>
</thead>
<tbody>
<tr>
<td>for the body</td>
<td></td>
<td></td>
</tr>
<tr>
<td>cotton overalls</td>
<td>yes</td>
<td></td>
</tr>
<tr>
<td>underwear</td>
<td>yes</td>
<td></td>
</tr>
<tr>
<td>Tyvek</td>
<td></td>
<td>yes</td>
</tr>
<tr>
<td>TOPSTRE yellow overalls</td>
<td>yes</td>
<td></td>
</tr>
<tr>
<td>WET overalls</td>
<td></td>
<td>yes</td>
</tr>
<tr>
<td>PE suit</td>
<td></td>
<td>yes</td>
</tr>
<tr>
<td>COOLINE cooling jacket</td>
<td>yes</td>
<td></td>
</tr>
<tr>
<td>for the head</td>
<td></td>
<td></td>
</tr>
<tr>
<td>cotton cap</td>
<td>yes</td>
<td></td>
</tr>
<tr>
<td>helmet</td>
<td>yes</td>
<td></td>
</tr>
<tr>
<td>for the hands</td>
<td></td>
<td></td>
</tr>
<tr>
<td>cotton gloves</td>
<td>yes</td>
<td></td>
</tr>
<tr>
<td>rubber gloves</td>
<td></td>
<td>yes</td>
</tr>
<tr>
<td>for the feet</td>
<td></td>
<td></td>
</tr>
<tr>
<td>protective shoes</td>
<td>yes</td>
<td></td>
</tr>
<tr>
<td>low covers</td>
<td></td>
<td>yes</td>
</tr>
<tr>
<td>high covers</td>
<td></td>
<td>yes</td>
</tr>
</tbody>
</table>

Figure 7.2 shows an example of additional protective equipment (Tyvek, respirator) and Table 7.5 gives the characteristics of various typical kinds of equipment.

**Respiratory protective equipment**

The basic aim of any type of breathing apparatus or **respirator** is to effectively reduce the inhalation of airborne substances which are harmful to health into the lungs. Since breathing equipment to a certain extent hampers natural breathing and/or the performance of work operations, decisions on the use of breathing equipment should be made in the knowledge that:
- Workers wearing a respirator carry out a given task 20 to 25 % more slowly than they would without a respirator. In an area where external radiation and air contamination are present, the decision should comply with the ALARA principle to minimize
the sum of the external and internal dose.
- Working with breathing equipment increases the load on the heart and lungs. For this reason all workers who are likely to use respirators must undergo a special additional medical examination.

Respiratory equipment falls into three main groups (Table 7.6):
- air-purifying respirators (air filtration),
- air-line respirators (with clean air supplied via a hose),
- self-contained breathing apparatus.

**Table 7.6:** Characteristics of respiratory equipment.

<table>
<thead>
<tr>
<th>type</th>
<th>shape</th>
<th>principle of operation</th>
<th>PF particles only</th>
<th>PF particles and gases</th>
</tr>
</thead>
<tbody>
<tr>
<td>air filtration</td>
<td>half facepiece</td>
<td>negative pressure during inhalation</td>
<td>10</td>
<td></td>
</tr>
<tr>
<td></td>
<td>full facepiece</td>
<td>negative pressure during inhalation</td>
<td>50</td>
<td></td>
</tr>
<tr>
<td></td>
<td>full facepiece</td>
<td>positive pressure</td>
<td>1000</td>
<td></td>
</tr>
<tr>
<td>air-line</td>
<td>half facepiece</td>
<td>continuous air flow</td>
<td>1000</td>
<td></td>
</tr>
<tr>
<td></td>
<td>half facepiece</td>
<td>flow during inhalation</td>
<td>10</td>
<td></td>
</tr>
<tr>
<td></td>
<td>full facepiece</td>
<td>continuous air flow</td>
<td>2000</td>
<td></td>
</tr>
<tr>
<td></td>
<td>full facepiece</td>
<td>flow during inhalation</td>
<td>50</td>
<td></td>
</tr>
<tr>
<td></td>
<td>full facepiece</td>
<td>positive pressure, flow during inhalation</td>
<td>2000</td>
<td></td>
</tr>
<tr>
<td></td>
<td>suit</td>
<td>continuous air flow</td>
<td>2000</td>
<td></td>
</tr>
<tr>
<td>self-contained breathing</td>
<td>full facepiece</td>
<td>flow during inhalation</td>
<td>50</td>
<td></td>
</tr>
<tr>
<td>apparatus</td>
<td>full facepiece</td>
<td>positive pressure, flow during inhalation</td>
<td>10000</td>
<td></td>
</tr>
<tr>
<td></td>
<td>full facepiece</td>
<td>closed-circuit breathing</td>
<td>50</td>
<td></td>
</tr>
<tr>
<td></td>
<td>full facepiece</td>
<td>closed-circuit breathing, positive pressure</td>
<td>5000</td>
<td></td>
</tr>
</tbody>
</table>

The first group includes e.g. a filter half mask (Figure 7.3), the second an air-line suit (Figure 7.4) and the third a mask with air supplied from a cylinder (Figure 7.5).
Figure 7.3: Filter half mask.

Figure 7.4: Protective clothing (suit) with an air line.

Figure 7.5: Face mask with air supplied from a cylinder.
Air-purifying respirators are able to reduce the concentration only of radionuclides bound to airborne dust particles or particulates (and some organic vapours), whereas respirators supplying clean air will also reduce the concentration of radionuclides in the gaseous state. The respirator's protection factor (PF) tells us by how much the concentration of activity in the inhaled air is lower than its concentration in ambient air:

\[
ac_{\text{inhaled}} = \frac{ac_{\text{ambient}}}{PF}
\]

A typical protection factor requirement for breathing equipment is that the concentration of activity in inhaled air is less than 0.1 DAC.

**Example:**
What should the protection factor of breathing equipment be for work in an area where the concentration of \(^{60}\text{Co}\) is 50 kBq/m\(^3\)?

**Answer:**

\[
PF = \frac{ac_{\text{ambient}}}{0.1 \text{ DAC}} = \frac{50 \cdot 10^3 \text{ Bq/m}^3}{0.1 \cdot 290 \text{ Bq/m}^3} = 1724 \approx 2000
\]

**Fit test**

A fit test is a procedure that tests the mask’s tightness or fit to the worker's face (Figure 7.6). There is a quantitative and a qualitative fit test.

**Figure 7.6:** Equipment for testing how well a mask fits to the face (fit test).
### 7.7 Exercises

1. Calculate the effective half-life for iodine $^{131}$I. Its physical half-life is 8.06 days and its biological half-life is 138 days.
2. Calculate the effective half-life for $^{137}$Cs. Its physical half-life is 30 days and its biological half-life is 70 days.
3. What dose rate is a worker exposed to when working unprotected in an atmosphere of 1 DAC of a given radionuclide?
4. Calculate the protection factor of respiratory equipment required for an atmosphere with 5000 Bq/m$^3$ of $^{131}$I. $\text{ALI}_{\text{inh}} = 2.6 \cdot 10^6$ Bq.
5. Calculate the protection factor of respiratory equipment required for an atmosphere with 3000 Bq/m$^3$ of $^{131}$I. $\text{ALI}_{\text{inh}} = 2.6 \cdot 10^6$ Bq, $V_{\text{inh}} = 2.4 \cdot 10^3$ m$^3$/year or 1.2 m$^3$/h.
8 RADIATION PROTECTION REGULATIONS

Learning objectives
After completing this chapter, the trainee will be able to:
1. Explain the aim of radiation protection.
2. Name the international organisations that issue recommendations and guidance on radiation protection.
3. Explain the basic ICRP principles.
4. Explain the concept of ALARA.
5. Give the dose limitations laid down by ICRP.
6. Identify different work site classifications and respective requirements.
7. Name the categories into which workers are classified.
8. Name the most important IAEA safety standard in the area of radiation protection.

8.1 The aim of radiation protection

People enjoy great benefits from the application of X-radiation, radionuclides and nuclear fuel in medicine, industry, research and electricity generation. However, working with sources of radiation by necessity implies a certain exposure of people to radiation, both during normal use and in potential accidents which no human activity is immune to. Under the assumption that any exposure of an individual to radiation poses a certain risk, exposure is only justified when the benefit derived from using a source is greater than this risk. Thus the fundamental principle of radiation protection is to assess the risks and benefits brought by the use of radiation. If the restrictions imposed on the use of radiations are too mild, the risks are unacceptably great; if the restrictions are too severe, the use of sources becomes too complicated or too expensive, depriving us of the benefits it could bring.

A simple example of assessing the positive and negative outcomes of using radiation is found in health care: a patient receives (compared to exposure in other activities) a relatively large dose, but the benefit of a radiological scan or therapy for the patient is greater than the radiation risk. Of course we should bear in mind that apart from the patients, exposure also affects health workers, although in a lesser degree. So it is also important to assess the benefit or risk brought by the use of radiation to society at large. The situation with radiation in nuclear power plants is similar. We know that nuclear power plants are very efficient and environmentally friendly energy facilities, and yet their operation unavoidably involves – relatively low, but nonetheless – exposure of workers to radiation, minimal discharges of radioactive substances, etc. This makes the question of benefits and risks a rather
complex one which can only be answered by society as a whole. Even in traffic where the risks are very obvious as well as easily proved, there are rules that strike a balance between benefits and risks: no doubt there would be fewer accidents if e.g. the speed limit on motorways was 30 km/h, but the general assessment the risk with a limit of 130 km/h is still acceptable and, at the same time, the benefit of faster-moving traffic is much greater than the increased risk of higher speeds.

8.2 International recommendations and standards

There are many international bodies and organisations around the world that study the effects of radiation and issue recommendations on radiation protection. They can be grouped into governmental and non-governmental organisations, or as those issuing merely non-binding recommendations and those international organisations which also have legislative power over their member states.

Non-governmental organisations include the ICRP and ICRU, which award membership to distinguished scientists based on their professional accomplishments.

International organisations which are either partly or fully focused on the issues of radiation protection and safety include the IAEA (International Atomic Energy Agency), the UNSCEAR (United Nations Scientific Committee on the Effects of Atomic Radiation), as well as the European Commission (EC). The UNSCEAR mostly deals with epidemiological studies of radiation effects and the IAEA issues safety standards which are aligned with ICRP recommendations and indirectly binding on member states (non-compliance leads e.g. to a withdrawal of IAEA’s technical support). On the other hand, EU directives are mandatory for member states.

ICRP recommendations

The ICRP (International Commission on Radiological Protection) is an independent, non-governmental professional body operating since 1928 and its primary goal is to issue recommendations on radiation protection.

As new findings about the effects of exposure to radiation came to light over the years, the ICRP repeatedly lowered the recommended annual dose limit (Figure 8.1).

Data on the increased number of cancerous diseases in atomic bomb victims clearly showed that, at high doses, the probability of developing radiation-induced cancer is proportional to the dose received. At low doses, the probability is so small that radiation-induced cancer cannot be statistically proved. However, there is also no firm evidence that sufficiently low doses are harmless, so the ICRP
adopted the conservative hypothesis of the linear dependence of the probability of stochastic effects with no threshold (the linear no-threshold model – LNT). This lead to a change in the overall concept of radiation protection: rather than compliance with the highest allowed doses, its goal is to reduce the probability of stochastic effects in the entire population, whether workers or inhabitants, to the lowest reasonable level.

Figure 8.1: The chronological development of the recommended annual dose limit for exposed workers (since 1960 also for the public).

The ICRP system of limiting radiation exposure is based on three principles:
- justification,
- optimization,
- dose limits.

Justification requires that sources of radiation be used only for those activities which bring more benefit than harm due to exposure to radiation.

Optimization requires that all exposures shall be kept as low as reasonably achievable (ALARA), economic and social factors being taken into account.

Dose limits are values of the effective dose or equivalent dose that should not be exceeded.

Today, the ICRP has an outstanding international reputation and authority. Despite having no legal force, its recommendations are applied by most countries in their national legislations.

IAEA Safety standards
The findings of the UNSCEAR and the recommendations of international expert bodies, notably the ICRP, are taken into account in developing the IAEA safety standards.
The preparation and review of the IAEA safety standards involves its own Secretariat and four safety standards committees for safety in the areas of nuclear safety (NUSSC), radiation safety (RASSC), the safety of radioactive waste (WASSC) and the safe transport of radioactive material (TRANSSC), and a Commission on Safety Standards (CSS) which oversees the programme for their development.

The principal users of IAEA safety standards in the member states are regulatory bodies and other relevant national authorities. All IAEA member states may nominate experts for the IAEA safety standards committees and may provide comments on draft standards. The CSS membership is appointed by the IAEA Director General and includes senior governmental officials having responsibility for establishing national regulations. A management system has been established for the processes of planning, developing, reviewing, revising and establishing the IAEA safety standards. It articulates the mandate of the IAEA, its vision on the future application of the IAEA safety standards, policies and strategies, and corresponding functions and responsibilities.

As indicated above, the ICRP is the main body which makes recommendations with respect to radiological protection matters in general, and dose limits in particular. For this reason, the IAEA has embodied the ICRP recommendations (i.e., those in its publication number 103 of 2007) in the latest publication of the **Basic Safety Standards** (Radiation Protection and Safety of Radiation Sources: International Basic Safety Standards, General Safety Requirements Part 3 of 2014 [8], hereafter referred to as **BSS**).

The BSS lay down requirements for establishing governmental, legal and regulatory frameworks for safety, specify basic obligations and administrative requirements for practices, dose limits (in accordance with the ICRP recommendations) and relevant radiation protection, management, technical and safety verification requirements. They also lay down detailed requirements for occupational exposure, public exposure, emergency exposure situations (including interventions), and also specify other exposures (potential, medical) to ionizing radiation. BSS also identify the protection of the environment as an issue necessitating assessment and the necessity of designing and implementing security and radiation safety measures in an integrated manner so that they do not conflict with each other.

Other publications of the IAEA safety standards series give more detailed guidance on how the BSS requirements should be met in particular situations. In this respect, Table 8.1 shows the current most relevant IAEA safety standards which can be used for achieving an appropriate level of safety and regulation of installations/facilities in the nuclear fuel cycle.
Table 8.1: Selected IAEA Safety standards that are most relevant to radiation protection.

<table>
<thead>
<tr>
<th>Type</th>
<th>Code no.</th>
<th>Title</th>
<th>Published</th>
</tr>
</thead>
<tbody>
<tr>
<td>Safety Requirements</td>
<td>GSR part 3</td>
<td>Radiation Protection and Safety of Radiation Sources: International Basic Safety Standards</td>
<td>2014</td>
</tr>
<tr>
<td>Safety Guide</td>
<td>RS-G-1.1</td>
<td>Occupational Radiation Protection</td>
<td>1999</td>
</tr>
<tr>
<td>Safety Guide</td>
<td>RS-G-1.2</td>
<td>Assessment of Occupational Exposure Due to Intakes of Radionuclides</td>
<td>1999</td>
</tr>
<tr>
<td>Safety Guide</td>
<td>RS-G-1.3</td>
<td>Assessment of Occupational Exposure Due to External Sources of Radiation Building Competence in Radiation Protection and the Safe Use of Radiation Sources</td>
<td>1999</td>
</tr>
<tr>
<td>Safety Guide</td>
<td>RS-G-1.4</td>
<td>Assessment of Occupational Exposure Due to External Sources of Radiation Building Competence in Radiation Protection and the Safe Use of Radiation Sources</td>
<td>2001</td>
</tr>
<tr>
<td>Safety Guide</td>
<td>RS-G-1.6</td>
<td>Occupational Radiation Protection in the Mining and Processing of Raw Materials</td>
<td>2004</td>
</tr>
<tr>
<td>Safety Guide</td>
<td>RS-G-1.8</td>
<td>Environmental and Source Monitoring for Purposes of Radiation Protection</td>
<td>2005</td>
</tr>
<tr>
<td>Safety Guide</td>
<td>RS-G-1.9</td>
<td>Categorization of Radioactive Sources</td>
<td>2005</td>
</tr>
</tbody>
</table>

For the purpose of establishing practical requirements for protection and safety, the standards distinguish between three different types of exposure situations: planned exposure situations, emergency exposure situations and existing exposure situations:

- A **planned exposure situation** is a situation of exposure that arises from the planned operation of a source or from a planned activity that results in an exposure from a source. Since provision for protection and safety can be made before embarking on the activity concerned, the associated exposures and their likelihood of occurrence can be restricted from the outset.

- An **emergency exposure situation** is a situation of exposure that arises as a result of an accident, a malicious act, or any other unexpected event, and requires prompt action in order to avoid or to reduce adverse consequences. Preventive actions and mitigatory actions have to be considered before an emergency exposure situation arises.

- An **existing exposure situation** is a situation of exposure which already exists when a decision on the need for control has to be taken. Existing exposure situations include situations of exposure to natural background radiation. They also include situations of exposure due to residual radioactive material that
derives from past practices that were not subject to regulatory control or that remains after an emergency exposure situation.

For each of these exposure situations requirements applicable to two categories of exposure are stated: for occupational exposures and for public exposures. Requirements for medical exposures are stated only for planned exposure situations.

For **planned exposure situations**, the following dose limits are set:

- For **occupational exposure of workers over the age of 18 years**, the dose limits are:
  (a) An effective dose of 20 mSv per year averaged over five consecutive years (100 mSv in 5 years), and of 50 mSv in any single year;
  (b) An equivalent dose to the lens of the eye of 20 mSv per year averaged over 5 consecutive years (100 mSv in 5 years) and of 50 mSv in any single year;
  (c) An equivalent dose to the extremities (hands and feet) or the skin of 500 mSv in a year.
  Additional restrictions apply to occupational exposure for a female worker who has notified pregnancy or is breast-feeding.

- For **occupational exposure of apprentices of 16 to 18 years of age** who are being trained for employment involving radiation and for exposure of students of age 16 to 18 who use sources in the course of their studies, the dose limits are:
  (a) An effective dose of 6 mSv in a year;
  (b) An equivalent dose to the lens of the eye of 20 mSv in a year;
  (c) An equivalent dose to the extremities (hands and feet) or the skin of 150 mSv in a year.

- For **public exposure**, the dose limits are:
  (a) An effective dose of 1 mSv in a year;
  (b) In special circumstances, a higher value of effective dose in a single year could apply, provided that the average effective dose over five consecutive years does not exceed 1 mSv per year;
  (c) An equivalent dose to the lens of the eye of 15 mSv in a year;
  (d) An equivalent dose to the skin of 50 mSv in a year.

For **emergency situations**, the Safety Standards specify generic criteria for acute doses for which protective actions and other response actions are expected to be undertaken under every circumstance to avoid or to minimize severe deterministic effects, and also sets guidance levels for restricting the exposure of emergency workers.

In BSS a number of requirements for government, regulatory bodies, registrants (registered users) and licensees, employers and workers are stated. We list some requirements related to the responsibilities of employers, registrants and licensees regarding protection of workers:
Employers, registrants and licensees shall be responsible for the protection of workers against occupational exposure. Employers, registrants and licensees shall ensure that protection and safety is optimized and that the dose limits for occupational exposure are not exceeded.

Employers, registrants and licensees shall establish and maintain organizational, procedural and technical arrangements for the designation of controlled and supervised areas, for local rules and for monitoring of the workplace, in a radiation protection programme for occupational exposure.

Registrants and licensees, in cooperation with employers where appropriate, shall establish, maintain and keep under review a programme for workplace monitoring under the supervision of a radiation protection officer or qualified expert.

Employers, registrants and licensees shall be responsible for making arrangements for the assessment and recording of occupational exposure and for workers’ health surveillance.

Employers, registrants and licensees shall provide workers with adequate information, instruction and training for protection and safety.

Employers, registrants and licensees shall not offer benefits as substitutes for measures for protection and safety.

Employers, registrants and licensees shall make special arrangements for female workers, as necessary, for protection of the embryo or foetus and of breast-fed infants. Employers, registrants and licensees shall make special arrangements for the protection and safety of persons under 18 years of age who are undergoing training.

The requirements for a legal and regulatory framework are supported by optimised criteria for exemption of practices and sources and by criteria for clearance of sources. The exemption level is a value established by a regulatory body and expressed in terms of activity concentration, total activity, dose rate or radiation energy, at or below which a source of radiation need not be subject to some or all aspects of regulatory control. The clearance level is a value established by a regulatory body and expressed in terms of activity concentration, at or below which regulatory control may be removed from a source of radiation within a notified or authorized practice.

A general requirement related to all aspects of radiation protection and safety is the requirement for a graded approach: the application of the Standards shall be commensurate with the characteristics of the practice or the source within a practice, and with the magnitude and likelihood of exposures.
8.3 Exercises and questions

1. What is the dose limit for the general public laid down by the BSS?
2. What are the prescribed dose limits according to the rules for occupationally exposed workers?
3. What does it mean if the activity of a nuclide is below the prescribed exemption level?
4. A radiography source of $^{60}$Co, is measured to have a dose rate of 4 mSv/h at a distance of five metres.
   a) What is the source activity?
   b) What is the dose rate at a distance of 2 m, at 0.5 m and at 10 cm?
   c) How long does it take to exceed a dose of 5 mSv at 2 m distance?
   d) How long does it take to exceed the annual limit on the equivalent dose to the skin for an occupationally exposed worker at a distance of 10 cm?
   e) How long does it take to exceed the annual dose limit for an occupational radiation worker at a distance of 2 m?
5. A dose rate of 10 mSv/h is measured at 1 m distance from a source of $\gamma$ radiation. At what distance from the source of radiation is the dose rate 10 $\mu$Sv/h? At what distance should the boundary of the controlled area be set?
9 RADIATION PROTECTION IN NUCLEAR INSTALLATIONS

Learning objectives

After completing this chapter, the trainee will be able to:

1. List some typical annual doses for workers in different stages of the nuclear fuel cycle.
2. Name the main radionuclides in the front-end of the nuclear fuel cycle and describe how they are formed.
3. Name the main families of radionuclides in a nuclear reactor and describe how they are formed.
4. Name at least two radionuclides from each family which are important from the perspective of radiation protection.
5. Describe the formation and main properties of hot particles.

9.1 Aspects of radiation protection in the nuclear fuel cycle

The nuclear fuel cycle contains a broad range of nuclear installations that employ a variety of technologies to process and utilize uranium, thorium and plutonium. These nuclear installations mainly include facilities for mining and milling, conversion, enrichment, fuel fabrication (including mixed oxide fuel - MOX), power and research reactors, interim spent fuel storage, reprocessing, waste disposal, and vitrification. Transportation systems are also recognized as an integral link between nuclear installations.

Table 9.1: Annual effective doses in the nuclear industry [15].

<table>
<thead>
<tr>
<th>Stage of nuclear fuel cycle</th>
<th>Dose (mSv)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Uranium mining</td>
<td>4.5</td>
</tr>
<tr>
<td>Uranium milling</td>
<td>3.3</td>
</tr>
<tr>
<td>Enrichment</td>
<td>0.1</td>
</tr>
<tr>
<td>Fuel fabrication</td>
<td>1.0</td>
</tr>
<tr>
<td>Nuclear reactors</td>
<td>1.4</td>
</tr>
<tr>
<td>Reprocessing</td>
<td>1.5</td>
</tr>
</tbody>
</table>

UNSCEAR has compiled data on doses received by about 800,000 workers in the nuclear industry worldwide [15]. In accordance with these estimates, the collective dose is about 1,400 man Sv and the average annual effective dose tends to be a little higher than 1 mSv. Nevertheless, doses have declined steeply in the last decade because of the widespread introduction of ICRP recommendations and the IAEA radiation safety standards (e.g. International Basic Safety Standards [8]). Table 9.1 shows the UNSCEAR estimates of the
average annual effective doses in relation to different occupations in the nuclear industry.

Criticality is one of the dominant safety hazards for non-reactor facilities, because they employ a great diversity of technologies and processes, and the materials of interest to radiation safety are more dispersed throughout them. They may be used not only in bulk form (fuel pellets, fuel elements, fuel rods, fuel assemblies, and so on), but in distributed and mobile forms as well (e.g., different kinds of solutions, slurries, gases, powders). This is in contrast to nuclear reactors, where the bulk of the nuclear material is located in the reactor core or fuel storage areas. As a result fissile materials may accumulate in some parts of the equipment and may also escape from the facility as a result of equipment leakage.

### 9.2 Important radionuclides for radiation protection

During the nuclear fuel cycle, different radionuclides are of major concern for radiation protection. At the front end of the fuel cycle (uranium/thorium mining and milling) only the natural radionuclides from the uranium (or thorium) decay series present a radiation risk. The resulting purified form of uranium concentrate is called “yellowcake” which is used in the preparation of fuel for nuclear reactors. It is denoted as U₃O₈ but is often actually ammonium diuranate [(NH₄)₂U₂O₇].

As yellowcake does not contain appreciable amounts of uranium progeny and uranium is very long-lived, the radiation risk is substantially lower in the next stages of fuel manufacturing (conversion to UF₆, enrichment, fuel fabrication). For MOX fuel fabrication, however, the radiation risk is substantially higher than for uranium fuel, because plutonium has a higher specific activity and radiotoxicity than uranium. At this stage, the main risk is from hot particles, very small particles (smaller than normal dust particles) containing actinides (uranium or plutonium) that may be inhaled.

Nuclear reactors are devices where the fission chain reaction takes place. The reactor core is therefore the main source of radiation during reactor operation. It is a very strong source of neutrons and fission products. Fission products are very highly radioactive while in the construction elements of the reactor neutrons produce additional sources of radioactivity, the so-called activation or corrosion products.

Nuclear reactors can be divided into power reactors (nuclear power plants) and research reactors. As power reactors generally have significantly higher power than research reactors, the amount of radioactivity produced is usually much larger than in the case of research reactors. However, the radiation protection issues are at least in principle similar for both types of reactors. Naturally, as NPPs are
much more complex installations than research reactors, the radiation protection problems are also much more complex.

In the back end of the nuclear fuel cycle (reprocessing and/or waste management, storage and disposal), all types of radionuclides (fission products, activation products, actinides) are important for radiation protection. As reprocessing or waste disposal usually takes place some time after the waste has been removed from the reactor and power plant systems, only the long-lived radionuclides (with half-lives comparable or greater than the time since removal from the reactor) are of importance.

**Natural radionuclides**

Figure 9.1 shows the decay chain of $^{238}\text{U}$. As uranium has a very long half-life, its daughter products (progeny) tend to be in radioactive equilibrium with uranium in soil or in ore. Radon-222, however, is an isotope of the noble gas radon and hence it is not chemically bound to soil but can easily diffuse to the surface.
**Figure 9.1:** The uranium-radium decay chain.

Radon and its short-lived daughter products are present in the air and are easily inhaled. Radon as inert gas is also exhaled. Its progeny are isotopes of polonium, bismuth and lead and they are chemically attached to dust particles in the air. They are adsorbed in the human respiratory system and remain there until they decay. In particular both polonium isotopes that emit alpha particles produce a substantial effective dose in the respiratory system. Radon therefore presents the largest radiation risk from natural radioactivity and, in fact, contributes one half of the average natural dose of population (1.2 mSv of 2.4 mSv total). A list of important radon daughter radionuclides with their properties is given in Table 9.2.

**Table 9.2:** Radon and its short-lived progeny.

<table>
<thead>
<tr>
<th>Nuclide</th>
<th>( t_{1/2} )</th>
<th>( E_{\alpha} ) [MeV]</th>
<th>( E_{\beta,\text{max}} ) [MeV]</th>
</tr>
</thead>
<tbody>
<tr>
<td>(^{222}\text{Rn})</td>
<td>3.8 d</td>
<td>5.5</td>
<td>-</td>
</tr>
<tr>
<td>(^{218}\text{Po})</td>
<td>3.1 min</td>
<td>6.0</td>
<td>-</td>
</tr>
<tr>
<td>(^{214}\text{Pb})</td>
<td>27 min</td>
<td>-</td>
<td>1.0</td>
</tr>
<tr>
<td>(^{214}\text{Bi})</td>
<td>20 min</td>
<td>-</td>
<td>3.2</td>
</tr>
<tr>
<td>(^{214}\text{Po})</td>
<td>164 µs</td>
<td>7.7</td>
<td>-</td>
</tr>
</tbody>
</table>

In uranium mines and milling facilities, the concentration of radon is substantially higher than the average natural background, therefore special measures need to be taken to protect the workers and the population in the neighbourhood of these facilities. These measures include ventilation and respiratory equipment.

**Fission products**

Fission products are a direct product of fission. They are produced within the reactor fuel and have by far the greatest activity of all radionuclides in the reactor or the nuclear power plant. During reactor operation or immediately after its shutdown, the total activity of a single fuel element is higher than \(10^{18}\) Bq, a month after shutdown it is around \(10^{17}\) Bq and after 10 years in the spent fuel pit it is around \(10^{16}\) Bq. Fortunately most fission products are elements which remain chemically bound in uranium dioxide and are consequently not released from the fuel pellet. In the context of nuclear safety, the pellet is thus considered as the **first barrier** that prevents the dispersal of radioactive materials.
Table 9.3: Major fission products with the main γ lines (they all emit β particles as well, but β energies are only given for pure beta emitters).

<table>
<thead>
<tr>
<th>Nuclide</th>
<th>$t_{1/2}$</th>
<th>$E_γ$ [MeV]</th>
<th>$E_{β, max}$ [MeV]</th>
</tr>
</thead>
<tbody>
<tr>
<td>$^{85m}$Kr</td>
<td>4.48 h</td>
<td>0.151</td>
<td></td>
</tr>
<tr>
<td>$^{133}$Xe</td>
<td>5.25 d</td>
<td>0.081</td>
<td></td>
</tr>
<tr>
<td>$^{133}$Xe</td>
<td>9.10 h</td>
<td>0.250</td>
<td></td>
</tr>
<tr>
<td>$^{131}$I</td>
<td>8.02 d</td>
<td>0.364; 0.637</td>
<td></td>
</tr>
<tr>
<td>$^{137}$Cs</td>
<td>30.2 y</td>
<td>0.662</td>
<td></td>
</tr>
<tr>
<td>$^{90}$Sr/$^{90}$Y</td>
<td>28.6 y</td>
<td>-</td>
<td>2.3</td>
</tr>
<tr>
<td>$^3$H</td>
<td>12.3 y</td>
<td>-</td>
<td>0.02</td>
</tr>
</tbody>
</table>

It is particularly the noble gases (Kr and Xe isotopes; practically entirely) which escape or diffuse out of the fuel pellet, as well as the isotopes of volatile elements (I, Cs, Sr, tritium; the bulk of the inventory). The pellet also partly releases the isotopes of Ba, Ce and Ru/Rh. The radionuclides which escape from the pellet build up in the empty space between the pellets and the fuel rod cladding, i.e. in the gap and plenum. The cladding is the second barrier to the dispersal of radioactive materials.

A PWR reactor core contains from 20,000 to 50,000 fuel rods and in a very small fraction of them the cladding does not stay perfectly tight throughout the service life. Some fission products (tritium and the noble gases in particular) even diffuse through the cladding. For this reason, a fraction of the fission products passes into the reactor coolant. Table 9.3 gives the key fission products. $^{133}$Xe has the greatest activity of all.

**Activation and corrosion products**

Neutrons also activate other materials in the core or its immediate environment: the coolant and the construction materials of the reactor. Such radionuclides are called activation products. The coolant that flows through the primary circuit (reactor, hot leg, steam generator, cold leg, reactor pump) causes corrosion and erosion processes on the walls of the primary circuit and on the construction elements of the core. As a result, tiny particles of these materials pass into the coolant and are activated as they travel through the core. Such radionuclides are called corrosion products.
Except in some special cases (working in the proximity of the reactor vessel, decommissioning) the most important activation products in terms of radiation protection are the radionuclides spread by the coolant through the primary circuit. These are either nuclides which form directly in the coolant or activated materials of e.g. the reactor vessel wall or core construction elements which enter the coolant due to erosion and/or corrosion. In nature, such material is not much different from the material entering the coolant by erosion-corrosion from other parts of the primary circuit, except that the materials may differ (the pressure vessel is made of steel and e.g. the steam generator tubes of nickel alloy). Activation and corrosion products are therefore discussed collectively. They typically spread throughout the entire primary system via the coolant and eventually a part of them binds to the inner walls of the pressure boundary. This process causes the contamination of all primary system components.

Besides activation and corrosion products, fission products are also released in the coolant. The pressure boundary of the primary circuit is thus the third barrier to the dispersal of radioactive materials. Table 9.4 gives the major activation and corrosion products.

In this table also the gases nitrogen in argon are listed. $^{16}\text{N}$ is formed by fast neutron reaction on oxygen in water molecules. It emits highly penetrating gamma radiation and is the most important source of radiation in the coolant during reactor operation. Argon is a component of air (its fraction in air is about 1%) and it is mostly formed in hollow spaces adjacent to the reactor vessel. In pool-type research reactors, air is present dissolved in the coolant and $^{41}\text{Ar}$ accounts for the largest part of the activity released.

**Hot particles**

The maintenance of pumps, valves and piping involves techniques such as cutting, grinding and welding which may produce tiny fragments that are carried away by the coolant. Often too small to be retained by filters, they are activated as they pass through the core. A similar thing happens to fragments resulting from the presence of...
foreign bodies in the primary system, which travel in water and cause damage to the walls and inner surfaces of primary system components. This results in microscopically small solid particles with high specific activity which are called hot particles.

Hot particles may contain a single radionuclide or a large number of radionuclides. These nuclides can be fission, activation or corrosion products. Exposure to hot particles is particularly significant in cases of a hot particle landing on the bare skin or protective clothing of workers, or their inhalation.

9.3 Questions

1. In what way do corrosion products form and what are their characteristics?
2. How does $^{60}$Co form in the primary system of a nuclear power plant?
3. Name two main sources of hot particles and three of their physical properties.
10 ENVIRONMENTAL MONITORING

Learning objectives
After completing this chapter, the trainee will be able to:
1. Explain the need for environmental monitoring.
2. Explain the meaning of the term “exposure pathway”.
3. Describe the main external exposure pathways for members of the public.
4. Describe the main internal exposure pathways for members of the public.
5. Explain the meaning of the term “constraint” for public exposure.
6. Explain the primary objectives of monitoring.
7. Explain the basis of environmental monitoring programmes
8. List some monitored constituents of airborne and liquid discharges.
9. Explain the terms representative person and critical group.
10. Explain the dose assessment approaches for members of the public.
11. State the main exposure pathways for different stages of the fuel cycle.
12. Compare doses from the natural background.

10.1 Need for monitoring nuclear facilities

In the current state of technology, the operation of nuclear and some radiation facilities would not be possible without some radioactive release to the environment. Almost all waste radioactive material is collected, treated and stored in the form of High Level Waste (HLW) or Low and Intermediate Level Waste (LILW), but waste processing technology cannot retain all the radioactive content within the facility boundaries, at least not at acceptable costs. Therefore small quantities of low specific activity liquids and/or gases are discharged into the environment. The consequence of these discharges is additional exposure of the population, primarily in the vicinity of the facility.

During normal operation discharges are subject to statutory control: they must be authorised and monitored. Regular monitoring of these controlled discharges, both at the source of the discharge and in the receiving environment, is essential in ensuring the protection of the public and the environment. Monitoring is generally carried out by the owners of nuclear and radiation facilities, and in some cases by regulatory agencies.

In radiation and nuclear safety the meaning of noun “monitoring” is not limited to the functions that have just been mentioned. According to International Basic Safety Standards [8], monitoring is “the measurement of dose, dose rate or activity related to the assessment or control of exposure to radiation or radioactive substances, and the interpretation of the results.” Depending on where the measurements are made, it could be classed as
individual (personal) monitoring, workplace monitoring, source monitoring and environmental monitoring. Depending on the purpose, it could be routine monitoring, task related monitoring and special monitoring.

During emergency situations releases are possible which highly exceed values in normal operation. While these releases will probably be uncontrolled, monitoring should be capable of assessing such elevated dose rates and activities in the environment to support an effective and timely emergency response.

### 10.2 Exposure pathways to the population

Controlled gaseous releases from nuclear facilities consist of a mixture of gases, vapours and particulates containing radionuclides which are usually discharged through the stack, although for small facilities they may be made through discharge vents or working hoods. Controlled liquid releases are typically contaminated water discharged via pipelines into rivers, lakes or the sea.

Gaseous releases spread in the form of a radioactive cloud, but a part of its radioactive content is also deposited on the ground and vegetation. This effect, called *fallout*, is enhanced with precipitation. Radionuclides from liquid discharge mix with water bodies and also in groundwater. Irrigation spreads these radionuclides to soil on cultivated land.

As a consequence of releases and radionuclide distribution in the environment, humans are exposed to radiation through different exposure pathways, which are presented in Figure 10.1.

![Figure 10.1: The possible pathways of exposure for members of the public as a result of discharges of radioactive material to the environment.](image)
environment [16].

The main external exposure pathways [16] are:

a) Source of radiation → humans: direct exposure from a source of ionizing radiation;

b) Source of radionuclides → atmosphere or water body → humans: exposure due to the plume of radionuclides in the atmosphere (‘cloud shine’) or in water;

c) Source of radionuclides → atmosphere or water body → human skin: contact exposure from radionuclides on the skin;

d) Source of radionuclides → atmosphere or water body → soil or sediment or building surface or vegetation → human: exposure from radionuclides deposited on the ground or on sediments (on the shores of rivers, lakes or the sea) or building surfaces (walls, roofs and floors) or vegetation (trees, bushes and grass).

The main internal exposure pathways [16] are:

a) Source of radionuclides → atmosphere → humans: inhalation of radionuclides in the plume;

b) Source of radionuclides → atmosphere or water body → (soil or sediment) → vegetation and/or meat, milk, eggs or marine food → human: ingestion of radionuclides in food or beverages;

c) Source of tritium → atmosphere → humans: for tritium oxide in the plume, by absorption through the skin;

d) Soil or sediment → humans: inhalation of resuspended radionuclides.

The importance of the various exposure pathways depends on:

- The radiological properties of the material released (e.g. gamma, beta or alpha emitters; physical half-life);
- The physical (e.g. gas, liquid or solid) and chemical (e.g. organic or inorganic form, oxidation state, speciation, etc.) properties of the material and its migration characteristics;
- The dispersal mechanism and factors affecting it (e.g. stack height, meteorological conditions, etc.) and environmental characteristics (e.g. climate, type of biota, agricultural production, etc.);
- The locations, ages, diets and habits of the exposed individuals or population.

As can be seen, the importance of each exposure pathway depends on the source properties, and the environmental and population characteristics. Therefore important exposure pathways must be identified and evaluated for each nuclear facility.

For all nuclear and radiation facilities identification and evaluation of exposure pathways must be done during the licensing process, when limits for discharges and dose constraint for public exposure are also laid down according to national legislation.
Constraint [8]:

A prospective and source-related value of individual dose (dose constraint) or risk (risk constraint) that is used in planned exposure situations as a parameter for the optimization of protection and safety for the source, and which serves as a boundary in defining the range of options in optimization.

For public exposure, the dose constraint is a source-related value established or approved by the government or the regulatory body, with account taken of the doses from planned operations of all sources under control. The dose constraint for each particular source is intended, among other things, to ensure that the sum of doses from planned operations for all sources under control remains within the dose limit.

Simply put, the dose constraint is the dose limit for members of the public related to a particular facility or source. Usually, dose constraints for nuclear facilities are from 0.1 mSv to 0.3 mSv per year.

10.3 Objectives of monitoring

The main objective of monitoring is verifying compliance with the requirements for protection and safety. As specified in IAEA Safety Standards [16], the primary objectives of any monitoring programme for protection of the public and the environment are to:

a) Verify compliance with authorized discharge limits and any other regulatory requirements concerning the impact on the public and the environment due to the normal operation of a practice or a source within a practice;

b) Provide information and data for dose assessment purposes, and to assess the exposure or potential exposure of populations due to the presence of radioactive materials or radiation fields in the environment from the normal operation of a practice or a source within a practice, and from accidents or past activities;

c) Check the conditions of operation and the adequacy of controls on discharges from the source and to provide a warning of unusual or unforeseen conditions and, where appropriate, to trigger a special environmental monitoring programme.

In summary, monitoring should be regarded as an essential element of the control of discharges to ensure protection of the public and the environment. It is also an essential element in determining the actions that should be taken to protect the public in intervention situations. The following three types of monitoring are envisaged [16]:

1) Source monitoring, which is monitoring the activity of
radioactive materials being released to the environment, or of external dose rates due to sources within a facility or activity;

2) **Environmental monitoring**, which is monitoring the external dose rates due to sources in the environment and/or the radionuclide concentrations in environmental media;

3) **Individual monitoring**, which is monitoring with equipment worn by individuals, or measurements of the quantities of radioactive materials in or on their bodies.

During normal operation of nuclear facilities, individual (personal) monitoring outside the facility is rarely justified, but in emergency situations it is required for emergency workers and also for public as appropriate. Source and environmental monitoring are performed routinely over the whole lifetime of the facility, and environmental monitoring even before commissioning (for assessment of pre-operational base-line conditions) and also after decommissioning of the facility.

### 10.4 Programmes for environmental monitoring

Environmental monitoring is implemented through programmes which must be based on the characteristics of the facility and an evaluation of the exposure pathways. The scale of a monitoring programme should be determined primarily by the significance of the expected doses to the public. The programme should be designed such that those radionuclides that are significant in terms of the dose received by a representative person are monitored.

The frequency of monitoring and sampling should be determined by the complexity of the environment, the significance of the doses to members of the public and the properties of the radionuclides. For example, if short lived $^{131}$I is to be monitored, the frequency of monitoring should be sufficient to detect it before it decays.

In the design of an environmental monitoring programme the following characteristics of the environment in the vicinity of the facility must be also taken into account:
(a) Prevailing wind direction;
(b) Meteorological variations;
(c) Current and future land use;
(d) Agricultural practices;
(e) Soil type and hydrological properties.

Relevant cultural, socioeconomic and demographic factors of the local population should be also considered in the design of the environmental monitoring programme.

Recommendations regarding the constituents and frequencies of sampling/measurement of radionuclides for normal operation is given
in Table 10.1. More information on programmes for environmental monitoring could be found in IAEA document SRS No. 64 [20].

**Table 10.1:** Environmentally monitored constituents and suggested frequencies of sampling and measurement for discharges of radionuclides to the environment [16].

<table>
<thead>
<tr>
<th>Discharge</th>
<th>Monitored constituents</th>
<th>Frequency</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>Airborne</strong></td>
<td><strong>External radiation</strong></td>
<td></td>
</tr>
<tr>
<td>Gamma dose rate</td>
<td>Continuously</td>
<td></td>
</tr>
<tr>
<td>Gamma dose — integrated</td>
<td>Twice a year</td>
<td></td>
</tr>
<tr>
<td>Neutron dose rate (if neutron radiation is foreseen)</td>
<td>Continuously</td>
<td></td>
</tr>
<tr>
<td>Neutron dose integrated (if neutron radiation is foreseen)</td>
<td>Twice a year</td>
<td></td>
</tr>
<tr>
<td><strong>Air, deposition</strong></td>
<td><strong>Continuously collection, weekly to monthly measurement</strong></td>
<td></td>
</tr>
<tr>
<td><strong>Foodstuff and/or ingestion</strong></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Leafy vegetables</td>
<td>Monthly during growing season</td>
<td></td>
</tr>
<tr>
<td>Other vegetables and fruits</td>
<td>Selected samples, at harvest</td>
<td></td>
</tr>
<tr>
<td>Grain</td>
<td>Selected samples, at harvest</td>
<td></td>
</tr>
<tr>
<td>Milk</td>
<td>Monthly when cows are on pasture</td>
<td></td>
</tr>
<tr>
<td>Meat</td>
<td>Selected samples, twice a year</td>
<td></td>
</tr>
<tr>
<td>Drinking water and/or groundwater</td>
<td>Twice a year</td>
<td></td>
</tr>
<tr>
<td><strong>Terrestrial indicators</strong></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Grass</td>
<td>Monthly when cattle are on pasture</td>
<td></td>
</tr>
<tr>
<td>Lichen, mosses, mushrooms</td>
<td>Selected samples, once a year</td>
<td></td>
</tr>
<tr>
<td><strong>Liquid</strong></td>
<td><strong>Aquatic dispersion</strong></td>
<td></td>
</tr>
<tr>
<td>Surface water</td>
<td>Continuous sampling, monthly measurement</td>
<td></td>
</tr>
<tr>
<td>Sediment</td>
<td>Once a year</td>
<td></td>
</tr>
<tr>
<td><strong>Aquatic foodstuffs</strong></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Fish</td>
<td>Selected samples, once a year</td>
<td></td>
</tr>
<tr>
<td>Shellfish</td>
<td>Selected samples, once a year</td>
<td></td>
</tr>
<tr>
<td><strong>Aquatic indicators</strong></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Seaweeds, marine sponges</td>
<td>Selected samples, once a year</td>
<td></td>
</tr>
<tr>
<td>Benthic animals</td>
<td>Selected samples, once a year</td>
<td></td>
</tr>
</tbody>
</table>

### 10.5 Assessment of doses to members of the public

Since the dose to the public cannot be measured directly, it must be estimated using environmental concentrations and appropriate habit data. Therefore, for the purpose of protection of the public, it is necessary to define a hypothetical exposed person to be used for
determining compliance with the dose constraint. This is called the representative person.

Current International Basic Safety Standards (BSS) [8] define a representative person as “the individual receiving a dose that is representative of the most highly exposed individuals in the population”.

In the previous version of International Basic Safety Standards [21] and in most national legislations dose constraints are related to members of “critical group”, where the critical group is “A group of members of the public which is reasonably homogeneous with respect to its exposure for a given radiation source and given exposure pathway and is typical of individuals receiving the highest effective dose or equivalent dose (as applicable) by the given exposure pathway from the given source” [21].

The problem with this definition in environmental monitoring is that neither is the critical group “critically” (i.e. badly or severely) exposed, nor is the dose constraint defined for the group (it relates to an individual). Another problem is that there are several critical groups, each related to a particular exposure pathway. It is possible that one group is critical for different (more than one) exposure pathways, but this still is not sufficient argument to use that particular group for comparison with dose constraint.

The concept of the critical group remains valid also in the current BSS considering that the dose to a representative person is the equivalent of, and replaces, the mean dose in the critical group. However, the concept of the representative person is not related to a particular exposure pathway and requires that all pathways are accounted for. Simple addition of doses from different critical groups is not possible, therefore groups should be also evaluated regarding other (noncritical) exposure pathways to identify the representative person.

Identification of critical groups for existing facilities, or a description of critical groups for future facilities is demanding and requires information about source characteristics and releases, environmental characteristics, population distribution and habits (e.g. amount of locally produced and consumed food). This information should be as realistic as possible. With this information and the results of measurements in the environment and on collected samples, it is possible to assess doses to critical groups.

In normal operation dose assessment is relatively straightforward when the external dose rate from a source or the concentration of radionuclides in the environment (also in food and water) exceeds the limits of detection for the measurement methods used and consumption data are available. Internal doses are calculated by
Multiplying the intake of radionuclides in air, water or food with the appropriate dose coefficients, which are age specific. It is recommended that doses are calculated for a 1-year-old infant, a 10-year-old child, and an adult representing the three age categories 0 to <6 years (infant), 6 to <16 years (child), and 16 to 70 years (adult).

When this is not the case, modelling must be used for dose estimation and verification of (undetectable) concentration levels. For this purpose additional data such as meteorological and hydrological conditions, land utilisation and population habits, characteristics of dwellings and occupancy should be available. If local data are not available, regional or generic estimates can be used, but this should be an exception, and not the rule.


Various computer codes are available for dose assessment, but codes and models should be verified and approved by the competent authority. Models and codes are very useful in prospective dose assessment, i.e. for doses that may be received in the future (e.g. next year, or during the operation period of a facility, or during facility decommissioning). This is also one of the most important parts of the environmental impact assessment process, which must be carried out during the stages of assessing the siting of a new nuclear programme [25].

**10.6 Exposure of the population from the various stages of the fuel cycle**

The fuel cycle comprises three main stages: fuel fabrication (including mining and milling), reactor operation and fuel reprocessing (including waste processing). At each stage of the nuclear fuel cycle, a variety of radionuclides are released in the form of liquids, gases, or solid particles. The nature of the effluent depends on the particular operation or process. Radionuclides discharged in all stages contribute to the effective dose of the population in the vicinity of the facilities, and also over a wider area. Long lived radionuclides are also distributed globally, contributing to the effective dose of all mankind. This contamination is not directly measurable globally and can be estimated only through modelling.
For all nuclear fuel cycle operations the local and regional exposures are estimated by UNSCEAR to be about 0.9 man Sv per gigawatt-year (GWy). The present world nuclear energy generation is about 250 GWy, so that the total collective dose from a year’s nuclear generation is about 225 man-Sv. Generally individual doses are low, being below 1 µSv in a year. However, certain individuals may receive higher doses because of where they live and what they eat.

In Table 10.2 the highest annual doses to the public due to discharges from the nuclear fuel cycle are listed. Also local and regional collective doses for commercial fuel cycle facilities up to year 1997 are given.

The releases of radioactive materials from conversion, enrichment, and fuel fabrication plants are generally small and consist mainly of uranium series isotopes. Inhalation is the most important exposure pathway.

For reactors, discharges of $^{14}$C, radioactive particulates, $^3$H (airborne and liquid), and noble gases radioisotopes are the main sources of doses. In the past, $^{131}$I was significant source of exposure, but now releases are minimal. Ingestion is the most important exposure pathway.

Doses from fuel reprocessing plants are mostly the consequence of discharges of $^{14}$C, $^{129}$I, $^{137}$Cs and $^3$H. In the past (before 1990), $^{137}$Cs was the most dominant radionuclide contributing almost 90% to the total collective dose (Table 10.2). The highest estimated doses were the consequence of local seafood consumption containing actinides. Also for other populations ingestion is the main exposure pathway.

**Table 10.2**: Annual doses due to discharges from the nuclear fuel cycle. (Data compiled from [15] and [26]).

<table>
<thead>
<tr>
<th>Stage of cycle</th>
<th>Type of effluent</th>
<th>Most exposed persons (mSv)</th>
<th>Local and regional collective dose to year 1997 (man Sv)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Fuel fabrication</td>
<td>Airborne</td>
<td>0.01</td>
<td>900</td>
</tr>
<tr>
<td></td>
<td>Liquid</td>
<td>0.01</td>
<td></td>
</tr>
<tr>
<td>Reactor operation</td>
<td>Airborne</td>
<td>0.001</td>
<td>2900</td>
</tr>
<tr>
<td></td>
<td>Liquid</td>
<td>0.004</td>
<td></td>
</tr>
<tr>
<td>Fuel reprocessing</td>
<td>Airborne</td>
<td>0.05</td>
<td>4700</td>
</tr>
<tr>
<td></td>
<td>Liquid</td>
<td>0.14</td>
<td></td>
</tr>
</tbody>
</table>
10.7 Natural background

As we have discussed in Chapter 1, there are a number of naturally occurring radionuclides in the environment. Some of them are also present in discharges from nuclear facilities (e.g. \(^3\)H, \(^{14}\)C, some members of the uranium series), while others are specific for the natural environment. Since natural radionuclides are present in air, soil, water, food, and all other commodities, they contribute significantly to public exposure. Inhalation of radon progeny, external gamma exposure to radionuclides in the soil and ingestion of (almost all natural) radionuclides are the main exposure pathways. In addition to external gamma exposure (which is approximately 0.06 µSv/h), people are also exposed to cosmic radiation. Exposure to cosmic radiation increases with altitude from approximately 0.03 µSv/h at sea level. For example, at 5 km it is 1 µSv/h and at 10 km it is 5 µSv/h.

Doses from the natural background are significantly higher than doses from nuclear facilities and dose constraints. Average doses from natural sources are presented in Table 10.3.

**Table 10.3:** Average radiation dose from natural sources [15].

<table>
<thead>
<tr>
<th>Source</th>
<th>Worldwide average annual effective dose (mSv)</th>
<th>Typical range (mSv)</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>External exposure</strong></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Cosmic rays</td>
<td>0.4</td>
<td>0.3 - 1.0(^a)</td>
</tr>
<tr>
<td>Terrestrial gamma rays</td>
<td>0.5</td>
<td>0.3 - 0.6(^b)</td>
</tr>
<tr>
<td><strong>Internal exposure</strong></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Inhalation (mainly radon)</td>
<td>1.2</td>
<td>0.2 - 10(^c)</td>
</tr>
<tr>
<td>Ingestion</td>
<td>0.3</td>
<td>0.2 - 0.8(^d)</td>
</tr>
<tr>
<td><strong>Total</strong></td>
<td>2.4</td>
<td>1 - 10</td>
</tr>
</tbody>
</table>

\(^{a}\) Range from sea level to high altitudes  
\(^{b}\) Depending on the radionuclide composition of soil and building materials  
\(^{c}\) Depending on indoor accumulation of radon gas  
\(^{d}\) Depending on the radionuclide composition of foods and drinking water

In addition to natural radionuclides and cosmic radiation, contamination from past atmospheric tests of nuclear weapons (global fallout) is still present globally and is also considered as a part of the natural background. The most important radionuclides from testing in terms of human exposure are now \(^{14}\)C, \(^{90}\)Sr and \(^{137}\)Cs. Ingestion is the most important exposure pathway, with a global average annual dose of 0.005 mSv (in 1963 the average annual dose was 0.1 mSv!). Fallout from past nuclear accidents (Chernobyl and Fukushima) is also considered as part of the natural background, although it is generally much lower and more or less limited to the northern hemisphere.
For new facilities, assessment of local values of global contamination is very important, since they are a part of the environmental baseline conditions.

**10.8 Monitoring in emergency exposure situations**

During an emergency situation there is a possibility of the elevated and uncontrolled release of radioactive material from a nuclear facility. The result of these releases would be additional doses to the population that will increase the likelihood of stochastic effects, or even doses high enough to give rise to severe deterministic effects. For some (smaller) releases doses will be insignificant and no special action is justified, but for other (higher) releases, the doses will exceed certain reference levels and protective and other response actions will be justified to avoid or mitigate the incidence of these doses.

Depending on the size, duration and characteristics of the release, the anticipated and possible development of source status, weather conditions and forecast, results of monitoring in the environment etc., the projected doses (or received doses in some circumstances) for the population demand one or more of the following protective actions: iodine thyroid blocking, evacuation, sheltering, prevention of inadvertent ingestion, decontamination of individuals, relocation, and restriction of consumption of food, milk and water. Generic criteria for acute doses for which protective actions and other response actions are expected to be taken under any circumstances to avoid or to minimize severe deterministic effects, and generic criteria for protective actions and other response actions to reduce the risk of stochastic effects are given in [8]. More information on protective actions and criteria that are used in emergency situations can be found in Module 16, Emergency Preparedness and Response.

To make sure that all protective actions during the emergency are implemented in time, emergency planning zones are defined around facility in the emergency response preparatory stage. These zones are defined according to the characteristics of required response in the most hazardous emergency situation possible in the nuclear facility.

For nuclear power plant accidents, emergency planning zones are defined as following:

1. Precautionary action zone (PAZ);
2. Urgent protective action planning zone (UPZ);
3. Extended planning distance (EPD); and
4. Ingestion and commodities planning distance (ICPD).

These zones encircle the nuclear power plant at approximately 3 to 5 km (PAZ), 15 to 30 km (UPZ), 50 to 100 km (EPD) and 100 to 300 km (ICPD). The PAZ is defined as the area where deterministic effects are possible in the most hazardous emergency situations,
therefore protective actions (e.g. evacuation) should be implemented before radioactive release from the power plant occurs. The UPZ is defined where exposure of the public could exceed reference levels in a short time (hours), and implementation of protective actions must start before or shortly after the release. The EPD and ICPD are zones where most protective actions are not implemented immediately, but only after measurement data from the environment are available.

Protective actions in the PAZ and UPZ should be initiated before or immediately after actual releases from the nuclear power plant start. The conditions in the power plant (emergency classification) represent the only initial information available. Later, during the development of the accident data from source monitoring (source releases) might be available (but not necessarily!), and later on, results of emergency monitoring from the environment should also become available to support initiation and implementation of effective protective actions. For planning purposes, the emergency monitoring programme is therefore divided into the following phases: the pre-release and early phase (release), the post-release or intermediate phase, and the recovery or remediation phase. Key data and requirements for environmental monitoring evolve with time and the transition from one phase to another.

**Objectives of emergency monitoring**

The specific objectives of emergency radiation monitoring in the environment are [16]:

a) To provide accurate and timely data on the level and degree of hazards resulting from a radiation emergency, in particular on the levels of radiation and environmental contamination with radionuclides;
b) To assist decision makers on the need to make interventions and take protective actions;
c) To provide information for the protection of emergency workers;
d) To provide information for the public on the degree of the hazard;
e) To provide information needed to identify any people for whom long term medical screening is warranted.

The emergency monitoring programme, which is a part of the emergency management system, must be designed for a range of possible emergencies. It must ensure that data all from instrumental measurements, sample collection and sample analysis, dose assessment and interpretation of results are available. For timely transfer of these data a proper communication system must be designed and established. The programme must also anticipate the receipt of assistance from other organisations or states, if needed.

**Source monitoring during an emergency**

Source monitoring is the only source of credible information in the
early phase of an emergency. Generally, it is mostly related to atmospheric, and sometimes also to aquatic releases.

The primary purpose of source monitoring is to determine the magnitude of the releases that might occur, that are occurring or that have occurred. Data on releases and meteorological data (also weather forecasts) serve as input to a computational model which can predict the spread of released radioactivity and estimate doses to the population. These predictions not only serve as the first orientation regarding any required protective actions, but can also provide valuable information for optimisation of the resources available for environmental monitoring during the emergency.

Source monitoring provides valuable information if the release takes place through a monitored path like the stack, or a discharge line. However, during an emergency unmonitored release paths are also possible. Therefore it is necessary to verify estimates from source monitoring data and identify possible additional points of release. This is especially important when release occurs as a consequence of some unanticipated event, such as an explosion. The most efficient approach in this case is to deploy properly trained and equipped personnel to assess the conditions.

Instruments for source monitoring must be designed for monitoring during an emergency. That means that they must have the appropriate dynamic range and must work reliably even in extreme environmental conditions (humidity, heat). Therefore a special accident monitoring system is used in nuclear facilities (see Module 15 In-plant accident management) which also provides information regarding source releases.

**Environmental monitoring during an emergency**

During the release, post-release and recovery phase of an emergency environmental monitoring is the most valuable source of data. It enables credible predictions of doses to the population and should be used as the decisive criterion for initiation and implementation of protective actions. However, elaborate dose assessment requires extensive monitoring data which is time consuming and so is not possible and acceptable in emergency conditions. Therefore Operational Intervention Levels (OILs) were developed, which serve as simple measurable criteria for when a particular protective action is justified. Examples of OILs for an accident in a light water nuclear power plant are the simple dose rate measured one metre above ground in the field, surface concentration of $^{137}\text{Cs}$ in the field, or an activity concentration of $^{137}\text{Cs}$ in food. OILs were developed on the assumption that the composition of release is similar irrespective of the size of the release. Default values of OILs are used in the initial phases of the emergency, and can be recalculated when accurate data on releases is available. More about OILs can be found in Module 16, Emergency preparedness and response.
In the release and post-release phase of an emergency, monitoring should enable identification of areas where a radioactive cloud is present and areas with significant contamination. These areas are identified by elevated gamma dose rates (not true if the release contains e.g. tritium or plutonium).

**Monitoring in the early phase of an emergency**

In the early phase of a severe accident involving airborne contamination the priorities for environmental measurements and sampling are as follows [16]:

1. Measurements of the external dose rate in air in defined zones around the plant to determine whether the OILs may be exceeded;
2. In-plume air sampling during the release to measure radionuclide concentrations for evaluation of inhalation hazards and revision of OILs;
3. After termination of the release, dose rate measurements over the area to identify places where OILs for evacuation, relocation, and restriction of food consumption are exceeded. Field gamma spectrometry should also be performed to assess the nuclide composition of ground contamination;
4. Specification of locations for continuous gamma measurement can be made for assessing gamma doses over extended periods;
5. Soil sampling and analysis for assessing ground deposition to supplement field gamma spectrometry measurements.
6. Sampling of contaminated food, milk and water after the end of release and plume passage enables a decision on food restrictions and possible food disposal to be made.

Collection and assessing these data requires activation of environmental monitoring teams. They are assembled and deployed to populated areas in the pre-release phase of an event. It is very important that teams are well trained and provided with necessary and reliable measuring equipment, means of communication and reliable maps. When support from other organisations or states is available, it is necessary to ensure in advance that results from different teams are compatible.

Measurements of external gamma dose and air sampling over large areas can be done very efficiently with an appropriately equipped plane or helicopter.

After plume passage samples of pasture, water, milk and other foodstuffs should be collected and measurements should be made to assess the need for restriction of their consumption. In nuclear accidents milk is especially important because of its content of iodine radioisotopes.
Monitoring in the post-release phase of an emergency

In the post-release phase local levels of deposited contamination should be evaluated in more detail. Field gamma spectrometry is the most useful method for assessing the conditions, but is demanding and time consuming. Data from gamma spectrometry should be supplemented by measurements of soil samples.

An extensive programme of sampling and measurement of vegetables and other locally grown produce, drinking water supplies and milk from local dairies is needed for comparison with the OILs. The extent and the nature of such sampling programmes will depend on the extent and the scale of the release and the demographics of the location in terms of local agricultural activities and the population distribution.

The public should be promptly provided with results of environmental monitoring or of other activities that directly involve them, their homes, their communities or their workplaces, as well as with interpretations of the results in terms of health risks and advice on urgent precautionary protective actions, and other response actions [16].

Personal monitoring

Personal (individual) monitoring of the population should be conducted together with environmental monitoring to determine whether decontamination or medical follow-up of people in the emergency zones is warranted. Monitoring is necessary when protective actions were delayed, i.e. evacuation was not finished before the beginning of release, or it is not possible to confirm this.

Monitoring of the population includes beta/gamma surveying at control points that must be established on the borders of the emergency zones. Personal monitoring of the public may also include measurements of internal contamination, either with a whole body counter, or by measurement of radionuclides in excreta, or just by non-spectrometric measurements of the concentration of $^{131}\text{I}$ in the thyroid gland, or of $^{134}\text{Cs}$ and $^{137}\text{Cs}$ in the whole body.

Personal monitoring of the population is practised rarely and may be appropriate only in condition of a severe emergency.

Personal monitoring of emergency workers includes measurement of external dose and measurement of external and internal contamination. Due to their tasks, the probability of incurring high doses and contamination is elevated. They should be provided with self-reading dosimeters and be informed about the site defined emergency worker turnback dose (EWTD). Methods of external and internal contamination assessment for emergency workers are the same as described above.
Data on personal monitoring of the population and emergency workers must be collected and evaluated, and should be used for medical follow-up of exposed individuals.

10.9 Questions

1. The most important characteristic of monitoring equipment is its sensitivity, since it will be used only for detection of low dose rates and low activity concentrations. Is this true and, if not, why?
2. Explain possible routes of population exposure when radioactive noble gas is discharged through the plant stack.
3. Explain possible routes of population exposure when radioactive particulates are discharged through the stack.
4. List some constituents that must be monitored for airborne discharges.
5. List some constituents that must be monitored for liquid discharges.
6. Explain the meaning of the term “representative person”.
7. Explain the meaning of the term “critical group”.
8. Which radionuclide is the highest contributor to population dose in the vicinity of a nuclear power plant?
9. Monitoring in the vicinity of a nuclear power plant has shown the presence of tritium (³H). What can you conclude about the operation of the power plant?
10. What are the specific objectives of emergency monitoring?
11. List the objectives of monitoring in different phases of an emergency.
11 REFERENCES

[14] INTERNATIONAL ATOMIC ENERGY AGENCY, Application of the Concepts of Exclusion, Exemption and


The views expressed in this document do not necessarily reflect the views of the European Commission.