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# **Practical Radiation Protection**

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**Tweede druk**

Oplage 2010

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Syntax Media – Arnhem

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This book is translated from Dutch. Original title *Praktische stralingshygiëne* published by Uitgeverij Syntax Media, Arnhem, 2008. ISBN 978 90 77423 61 5.

ISBN 978 90 77423 63 9

[www.syntaxmedia.nl](http://www.syntaxmedia.nl)

Ontwerp omslag: Lapis Vivus grafisch ontwerp, Oosterbeek

Opmaak: AlphaZet prepress, Waddinxveen

## Chapter 1

# Structure of the atom and decay

### 1-1 Structure of an atom

All matter is composed of atoms. Each atom is made up of a positively charged nucleus and a surrounding ‘cloud’ of moving electrons. The electrons have little mass as compared with the nucleus, i.e. the mass of the atom is concentrated almost entirely in the nucleus. In the case of an electrically neutral atom, the mass of the electron cloud is about 1/4000-th the mass of the nucleus. The nucleus contains two types of particles: protons and neutrons. Protons carry a positive charge. Neutrons are particles without charge. The mass of a neutron is almost identical to that of a proton. The letter Z, the atomic number, denotes the number of protons. The letter N, the neutron number, denotes the number of neutrons. Because the mass is determined almost exclusively by protons and neutrons, the mass number is derived from the sum:  $A = Z + N$ .

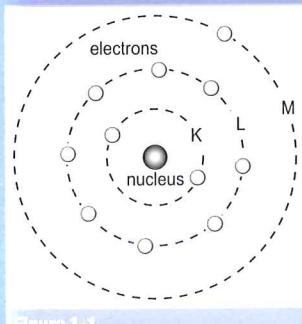


Figure 1-1

The Na-atom.

In electrically neutral atoms, the number of electrons is equal to Z and independent of N. The electrons are ordered in shells. Every shell contains only a finite number of electrons. The shells are specified, beginning with the nucleus, with the letters K, L, M, N etc. (see figure 1-1).

The electrons in a shell that is far away from the nucleus are shielded against the positively charged nucleus by the electrons in the inner shells, and are therefore loosely bound. The outermost electrons determine the chemical properties of an element (or atom). All types of atoms that have the same Z but different N have the same number of outermost electrons and therefore the same chemical properties; they are called isotopes. Types of atoms that have different Z and different N but have the same mass number A are called isobars. The energy content of nuclei with the same composition can vary. These nuclei can be in an excited state, in which case they are called isomers. The various atoms of all elements are referred to as nuclides. For a brief unambiguously description of a nuclide, one uses the notation  ${}^A_Z X$ , where the superscript A is the mass number, the subscript Z is the atomic number and X is the atom under consideration. If X is named, Z can be omitted.

**Example**

Copper occurs in nature in two isotopic forms. Since  $Z=29$  for copper one can refer to these isotopes as  $^{63}\text{Cu}$  and  $^{65}\text{Cu}$ . The complete expression for each is  $^{63}_{29}\text{Cu}$  and  $^{65}_{29}\text{Cu}$  respectively. Cu-63 and Cu-65 are alternative notations.

## 1-2 Stability of atomic nuclei

In theory, by varying  $Z$  and  $N$  an infinite number of combinations can be formed. However, only a limited number of combinations (about 1900) are known. The maximum value for  $Z$  is 107 and that for  $N$ , 159. Of the various possibilities, only 267 turn out to be stable, the remainder being unstable. Nuclides with unstable combinations decay emitting high energy radiation and are called *radionuclides* or *radioisotopes*. The stability depends mainly on the  $N/Z$ -ratio.

Figure 1-2 is a graphical illustration in which  $Z$ , the number of protons, is plotted on the vertical axis and  $N$ , the number of neutrons, on the horizontal axis. The stable nuclides are indicated by closed squares and the unstable ones by open squares. The stability depends mainly on the  $N/Z$  ratio; for small values of  $Z$ , the condition of stability is satisfied for a nucleus in which  $Z$  is equal (or almost equal) to  $N$ . For large values of  $Z$ , there tend to be more neutrons than protons to compensate for the repulsive forces between the protons, and the  $N/Z$  ratio rises to about 1.5. An even number of protons and neutrons also gives rise to more stability. Nuclear species with the same mass num-

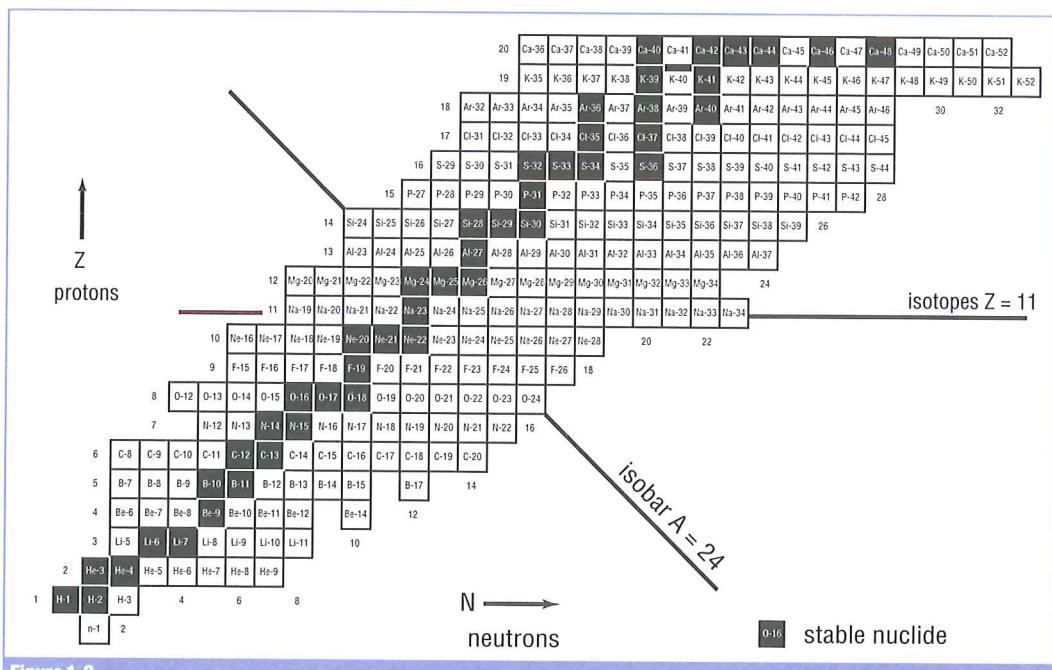


Figure 1-2

The chart of nuclides.

ber  $A$  are called isobars and appear along lines drawn at  $45^\circ$ . The  $N/Z$  graph discussed above provides a good overview of stable and unstable nuclides. The graph is available in a variety of forms and is called the isotope chart or *chart of the nuclides* (e.g. the chart of the nuclides of the Nuclear Research Centre in Karlsruhe).

### 1-3 Radionuclides

#### *Natural radioactivity*

Eighty-one different stable elements occur in nature. The lightest is hydrogen with  $Z = 1$ . Bismuth has the highest  $Z$ , namely 83. The elements missing from the series of stable elements are technetium (Tc) with  $Z = 43$ , and promethium (Pm) with  $Z = 61$ . In addition to these stable elements, some elements exist in nature only as radionuclides:

- thorium, in the form of  $^{232}\text{Th}$
- uranium, with three isotopes  $^{234}\text{U}$ ,  $^{235}\text{U}$  and  $^{238}\text{U}$
- radium ( $^{226}\text{Ra}$ ), decay product of  $^{238}\text{U}$
- radon ( $^{222}\text{Rn}$ ), as a daughter of  $^{226}\text{Ra}$ .

In nature, there are also stable elements with a very minute fraction of radioactive isotopes (radioisotopes). Examples of these are:

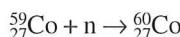
- hydrogen, H, with the isotope  $^3\text{H}$ , called tritium
- carbon, C, with the isotope  $^{14}\text{C}$
- potassium, K, with 0.0117%  $^{40}\text{K}$ .

$^{40}\text{K}$ , thorium and uranium have existed in the earth crust since the early formation. The isotopes  $^{14}\text{C}$  and  $^3\text{H}$  are constantly formed in the upper regions of the earth's atmosphere in nuclear reactions caused by cosmic radiation.

#### *Artificial radioactivity*

In addition to the stable and the radioactive nuclides that occur in nature, there are many hundreds of artificially made radioactive nuclides. They are made in particle accelerators and nuclear reactors. In these accelerators, nuclei are bombarded with particles such as electrons or protons or with even heavier ionised particles. These are often combinations of protons and neutrons (in other words nuclei) such as deuterons ( $^2\text{H}$ ), tritons ( $^3\text{H}$ ) and helium nuclei ( $^4\text{He}$ ).

Many types of radionuclides are also formed in nuclear reactors by the fission of the heavy nuclei of uranium, thorium or plutonium. Because neutrons are released during this fission, irradiating stable nuclei with these neutrons can produce all kinds of radionuclides. For instance, irradiating stable cobalt with neutrons produces the radioactive isotope of cobalt.



${}_{27}^{60}\text{Co}$  decays slowly, producing the stable nickel isotope  ${}_{28}^{60}\text{Ni}$  and releasing radiation.

## 1-4 Decay and the law of decay

Unstable nuclides decay spontaneously to another nuclide. Spontaneity is a fundamental property of *radioactive decay*. It is impossible to predict when the nucleus of an unstable nuclide will disintegrate; you can compare it to throwing a dice: you cannot tell whether you will throw a six next. However, the disintegration process can be described mathematically when there are many nuclei. The number of disintegrations per unit time, thus the decrease of the number of radioactive nuclei, is called the *activity*. This activity is proportional to the number of the existing nuclei.

The activity  $A(t)$  at time  $t$  is described as

$$A(t) \stackrel{\text{def}}{=} -\frac{dN(t)}{dt} = \lambda \cdot N(t) \quad (1-1)$$

where

$\frac{dN(t)}{dt}$  = the change in the number of nuclei over time

$\lambda$  = decay constant

$t$  = time

The negative sign indicates that the number of radioactive nuclei is decreasing.

At  $t = t_0$ ,  $N = N_0$  and  $A = A_0$ , the solution of the simple differential equation (1-1) is:

$$N(t) = N(0) \cdot e^{-\lambda \cdot t} \quad (1-2)$$

from which follows that equation (1-1) can be written as:

$$A(t) = A(0) \cdot e^{-\lambda \cdot t} \quad (1-3)$$

The time it takes for the number of active nuclei of a certain nuclide to be halved is a constant,  $T_{1/2}$ , the *half-life*.

The fact that this is a constant can be seen as follows. Let  $\Delta t$  be the time it takes for the activity  $A(t)$  to be halved. Then according to (1-3):

$$\frac{A(t + \Delta t)}{A(t)} = \frac{1}{2} = e^{-\lambda \cdot \Delta t}$$

If we take the natural logarithm:

$$\ln \frac{1}{2} = -\ln 2 = -\lambda \cdot \Delta t \quad \Rightarrow \quad \Delta t = \frac{\ln 2}{\lambda}$$

We see now that  $\Delta t$  – the time it takes to halve the number of nuclei, and of the activity – is independent of the original activity and that  $\Delta t (= T_{1/2})$  is only dependent of  $\lambda$ .

$$\Delta t = T_{1/2} = \frac{\ln 2}{\lambda} \quad \text{so} \quad \lambda = \frac{\ln 2}{T_{1/2}} \quad (1-4)$$

By substituting the result of (1-4) into (1-3), one obtains a frequently used form of the law of decay:

$$A(t) = A(0) \cdot e^{-\frac{\ln 2}{T_{1/2}} \cdot t} \quad (1-5)$$

This is the same as:

$$A(t) = A(0) \cdot \left(\frac{1}{2}\right)^{\frac{t}{T_{1/2}}} \quad (1-6)$$

The decay of a radionuclide is shown in figures 1-3 (linear plot of the activity versus time) and 1-4 (semi-logarithmic plot). In the latter case, the relationship is linear, as expected.

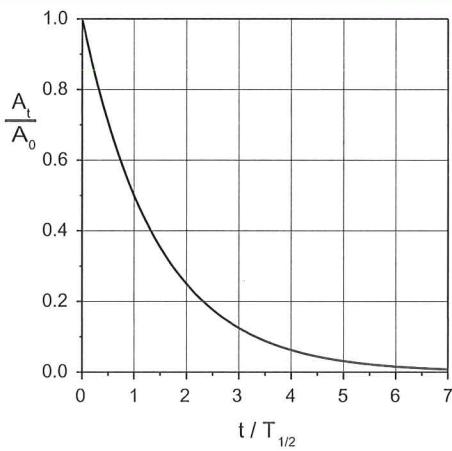


Figure 1-3

A linear plot of the kinetics of the decay process; the activity is shown on the vertical axis, and the time is shown on the horizontal axis.

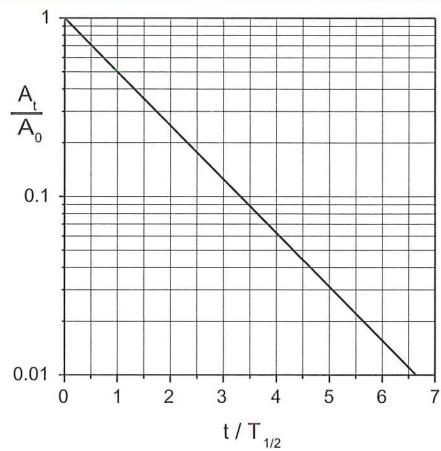


Figure 1-4

The same information as in figure 1-3 plotted on a semi-logarithmic scale.

## 1-5 Quantities and units of activity and energy

### *Unit of activity*

The internationally accepted unit of activity of a radioactive substance is the *becquerel* (Bq), named after the French scientist Antoine Henri Becquerel who discovered radioactivity. It is simply the number of disintegrations per second (dps). The dimension of Bq is  $\text{time}^{-1}$ , with the unit  $\text{s}^{-1}$ . The old unit, sometimes still used, is the *curie* (Ci), named after the famous husband and wife team of researchers, Marie and Pierre Curie. This unit is much larger than the becquerel:

$$1 \text{ Ci} = 3.7 \cdot 10^{10} \text{ Bq}$$

Initially the curie was assumed to represent the activity of 1 gram of radium. In practice, the becquerel and the curie are not always convenient to use. 1 Bq is very little activity. Therefore, the following prefixes are used: k (kilo), M (mega) and G (giga):

- $1 \text{ kBq} = 10^3 \text{ Bq}$
- $1 \text{ MBq} = 10^6 \text{ Bq}$
- $1 \text{ GBq} = 10^9 \text{ Bq}$ .

The *specific activity* of a nuclide is the activity per unit of mass of the substance. If a substance has high activity, this does not necessarily mean that it has a large mass. 37 MBq (1 mCi) of radium ( $^{226}\text{Ra}$ ,  $T_{1/2} = 1600$  years) is equivalent to about 1 milligram of radium. For a substance with a short half-life, e.g.  $^{123}\text{I}$ , 37 MBq is equivalent to 0.000001 milligram.

### *Unit of energy*

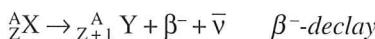
Usually, energy is expressed in the unit joule (J). For energies on the atomic scale – such as binding energy between two atoms in a molecule – or for energies that play a role in nuclear processes, this unit of measure is too large. A more useful unit for these purposes is the electron volt (eV). This is the kinetic energy an electron acquires traversing a potential difference of 1 volt. One eV is a very small amount of energy,  $1.6 \cdot 10^{-19} \text{ J}$ , but it is very useful to express the above mentioned energies. The binding energy in molecules is of the order of a few eV's. The energies that play a role in nuclear processes are three to six orders of magnitude higher ( $10^3$  to  $10^6$ ) and are expressed as keV or MeV. From this difference between binding energies and the energy that is released during nuclear processes, we can conclude that nuclear processes are capable of breaking the bindings in molecules.

## 1-6 Modes and schemes of decay

The composition and the energy state of the nucleus determine the stability or instability and also the mode of decay. Nuclei can have an excess of protons, neutrons or energy. Each state leads to a specific mode of decay. In a decay process, there is always a transition from a higher to a lower energy level. This difference in levels varies from keV's to MeV's.

### 1-6-1 Modes of decay when there is an excess of neutrons

A nucleus with an excess of neutrons can reach a (more) stable state by emitting a neutron. This type of reaction, however, is rare and is only relevant for nuclei with a large energy surplus. The only really important reaction is that a nucleus emits a  $\beta^-$ -particle (accompanied by a anti-neutrino =  $\bar{\nu}$ ), whereby a neutron changes into a proton in the nucleus:



An example is:

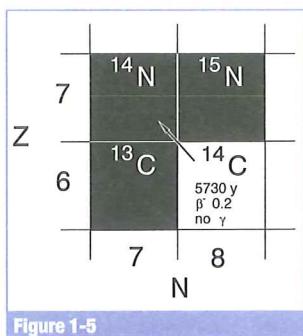
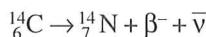


Figure 1-5

$\beta^-$ -decay on the nuclide chart.

A  $\beta^-$ -particle is physically identical to an electron; it is only called a  $\beta^-$ -particle when it is emitted by the nucleus. Since a neutron has changed into a proton the mass number  $A$  does not change. On the chart of the nuclides,  ${}^A\text{X}$  and  ${}^A\text{Y}$  also lie on one isobar, diagonally side by side. See figure 1-5. The  $\beta^-$ -particle is in fact the radiation that makes the decay-process detectable. Because the decay-energy is divided between the  $\beta$ -particle and the anti-neutrino the  $\beta$ -particle has an energy that varies from zero to a maximum,  $E_{\beta,\max}$ .

$$E_{\beta^-} = E_{\beta,\max} - E_{\bar{\nu}}$$

The mean value of the energy of a  $\beta$ -particle is roughly  $1/3 E_{\beta,\max}$ . The neutrino hardly interacts at all with matter and is therefore almost undetectable and is unimportant in radiation protection.

### 1-6-2 Modes of decay when there is a shortage of neutrons

When there is a shortage of neutrons - this is equivalent to a surplus of protons - there is a possibility that a proton will be emitted. However, this reaction hardly ever occurs. Two reactions that do regularly occur are electron capture and positron emission.

#### Electron capture

The nucleus captures an electron from the electron cloud; this electron usually comes from the innermost (K-) shell; consequently, a proton in the nucleus transforms into a neutron (and a neutrino is released);

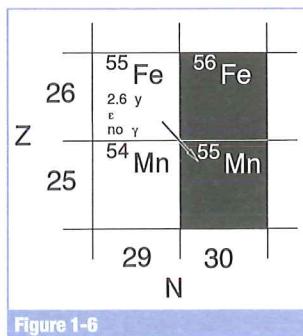
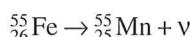


Figure 1-6

Electron capture.



An example is:



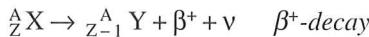
As a result, a vacancy arises in the electron shell. An electron from one of the outer shells immediately occupies this vacancy, which results in the emission of energy in the form of electromagnetic radiation. This energy is relatively high and the photon that arises is very penetrating. Electromagnetic radiation that is generated during these atomic processes, in which electrons change their course (jumps between orbits) is called characteristic X-radiation. This radiation makes the decay detectable and causes the radiation risk.

In addition to X-rays, electrons in the outer shells, which are loosely bound, are ejected. These ejected electrons are called Auger electrons. They generally are of low energy and are therefore difficult to detect.

### Positron emission

If the nucleus emits a  $\beta^+$ -particle (accompanied by a neutrino), this is called  $\beta^+$ -emission. This particle, which is called a positron, is an electron with a positive charge.

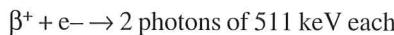
It is an anti-particle, a kind of mirror image of an electron. The reaction is:



An example is:



Just like a  $\beta^-$ -particle, the  $\beta^+$ -particle is easily detectable and its energy can vary from zero to  $E_{\beta,\max}$  if the  $\beta^+$ -particle comes to rest after a number of interactions; it unites with an electron. The entire mass will convert to energy in the form of electromagnetic radiation (photons)

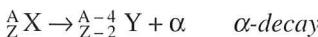


This process is known as *annihilation*. These photons of 511 keV are characteristic for annihilation radiation, and as a consequence for  $\beta^+$ -radiation.

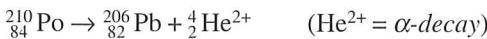
In this decay process a total energy of  $E_{\beta,\max} + 2 \cdot 511 \text{ keV}$  is released. It can be proven that  $\beta^+$ -emission is only possible if the energy difference between the beginning and the end-state is at least  $2 \cdot 511 = 1022 \text{ keV}$ . Since neither electron capture nor  $\beta^+$ -emission changes the mass number of the nucleus, the nuclei  ${}^AX$  and  ${}^AY$  are on an isobar, on the chart of the nuclides (figure 1-6 and 1-7).

### 1-6-3 $\alpha$ -decay

Heavy nuclei, particularly those with an excess of protons, become stable by emitting a particle consisting of two protons and two neutrons, the  $\alpha$ -particle. In fact, this is a helium nucleus:  $\text{He}^{2+}$ . This reaction can be written as follows:



An example is:



In radionuclide laboratories,  $\alpha$ -emitters are seldom used. Working with  $\alpha$ -emitters involves more risks than with  $\beta$ - or  $\gamma$ -emitters. This is because  $\alpha$ -emitters have a considerable *recoil effect* due to the relatively large mass of the  $\alpha$ -particle; as a result  $\alpha$ -emitters can disperse spontaneously, causing an extra risk of internal contamination. In addition,  $\alpha$ -emitters as a rule are far more radiotoxic than  $\beta$ - or  $\gamma$ -emit-

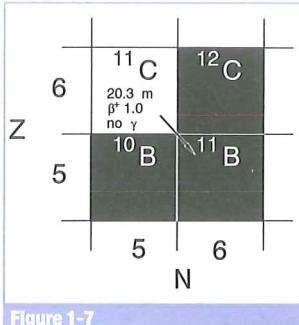


Figure 1-7

$\beta^+$ -decay.

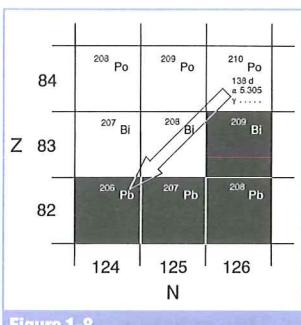


Figure 1-8

$\alpha$ -decay.

ters. On the chart of the nuclides X and Y are separated by two squares and lie on a line perpendicular to an isobar. See figure 1-8.

#### 1-6-4 $\gamma$ -decay, internal conversion

When a nucleus has decayed by one of the methods mentioned above it is not always immediately stable or in its ground state. The  $\alpha$ - or  $\beta$ -decay is generally followed by the emission of electromagnetic radiation, which removes the excess energy. This form of radiation is called  $\gamma$ -radiation and the process is called  $\gamma$ -decay. A photon, the  $\gamma$ -photon, is emitted from the nucleus. Generally this  $\gamma$ -radiation is easy to detect, due to the energy that is usually rather high.

Gamma radiation, X-rays and light are electromagnetic radiation. They are discriminated on the basis of their origination:

- Gamma radiation originates from the nucleus.
- X-rays originate from extra-nuclear processes and are produced by energy loss of electrons by slowing down, or by a transition from one atomic orbit to another with a lower energy. Often, but not always, X-rays have a lower energy than  $\gamma$ -radiation.
- Light is also produced by a transition of an electron from one orbit to another, but in this case it concerns the outermost orbits, with low energies. This emission of light is called 'scintillation'.

So X-rays and visible light are both produced by a change of orbit of electrons; only when this radiation has penetrating and ionising properties we call this radiation X-rays. X-rays have a high energy compared to light.

During  $\gamma$ -decay the composition of the nucleus does not change ( $Z$  and  $N$  remain constant). A nucleus can have various energy states (excited states) above its ground state; these are referred to as  $m_1$ ,  $m_2$ , .... This distinction only makes sense when the half-life is not too short, for instance longer than 0.1 second. The highest index is used for the highest energy. Because the composition of the nucleus remains unchanged ( $Z$  and  $N$  remain constant) such nuclides, i.e. nuclides of which only the energycontent changes, are called isomers. For example,  $^{137m}\text{Ba}$  is an isomer of the stable  $^{137}\text{Ba}$ .

Instead of  $\gamma$ -decay, another process, called *internal conversion (I.C.)*, can occur. This process involves the direct transfer of the energy of the nucleus to an electron, generally an electron from the innermost shell. This electron is thereby ejected from its orbit with large energy and can be detected in the same manner as a  $\beta$ -particle. It leaves a vacancy, which is readily occupied accompanied by the emission of X-rays, which are also detectable. These electrons are called conversion electrons. They have a sharply defined energy, in contrast to the  $\beta$ -particles that share their energy with neutrinos. The energy of the conversion electron is in the same order of magnitude as the energy of the  $\beta$ -particles, so the radiation from nuclides that emit conversion electrons is comparable to the radiation of  $\beta$ -emitters.

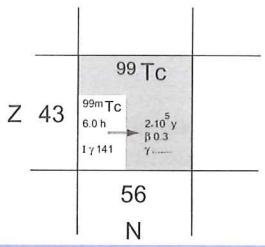


Figure 1-9

Isomers.

On the chart of the nuclides, isomers are depicted up in one square! See figure 1-9.

In some cases, an isomer does not return from the excited state to the ground state, for example,  $^{71}\text{m}\text{Zn}$  does not decay to  $^{71}\text{Zn}$  via  $\gamma$ -decay, but decays to  $^{71}\text{Ga}$  via  $\beta^-$ ,  $\gamma$ -decay.

A process that is similar to internal conversion is the origination of *Auger electrons*. The energy that can originate as X-rays when a vacancy in an innermost shell is occupied, can also be transferred to an electron in a shell that is more to the outside. That electron is thereby ejected from its orbit, just like the conversion electron. These electrons have relatively low energies and can contribute to the radiation dose.

### 1-6-5 Spontaneous fission

Very heavy nuclei can undergo *spontaneous fission* into two fragments, releasing 2 to 3 neutrons. The mass numbers of these fragments are about 95 and 140. The energy released during this process (in the form of kinetic energy of the particles) is high, about 200 MeV.

### 1-6-6 Decay-schemes

A *decay-scheme* shows the decay in a graph. The energy is plotted on the vertical axis. In this scheme, the top level is called the parent level; below that lies the daughter level. The difference between these two levels is the disintegration energy (i.e. the energy released in the decay) and is usually denoted by the letter  $Q$ : the  *$Q$ -value*. An arrow slanted to the bottom right denotes an increase in the Z-value, one pointing downwards to the left a decrease in the Z-value, see figure 1-10.  $\alpha$ -decay is depicted by an ‘two-line’ bent arrow (2 n and 2 p),  $\beta^+$ -decay by a bent arrow because for this process an energy,  $Q$ , of more than 1022 keV is needed.

In practice, decay processes are often much more complicated.  $\beta$ -decay often occurs via several routes, each with its own probability.  $\beta^+$ -decay is nearly always accompanied by electron capture and vice-versa. One must therefore take the emission of X-rays into account. Only if the energy difference between parent and daughter is less than 1022 keV, electron capture is not accompanied by  $\beta^+$ -emission. In addition,  $\beta$ - and  $\alpha$ -decay are often followed by  $\gamma$ -decay. If the latter is prompt, it is called coincidence; if it is not direct it is called an isomer, see section 1-6-4.

### 1-6-7 Parent-daughter relations

After a (parent) nuclide has decayed, the newly formed (daughter) nuclide may also be unstable. If the daughter nuclide has a short half-life compared to that of the parent, then an *ingrowth* of activity occurs, during which the total activity grows.

Assume for instance that during a certain period of time 100 parent nuclei disintegrate. If the daughter decays much faster compared to

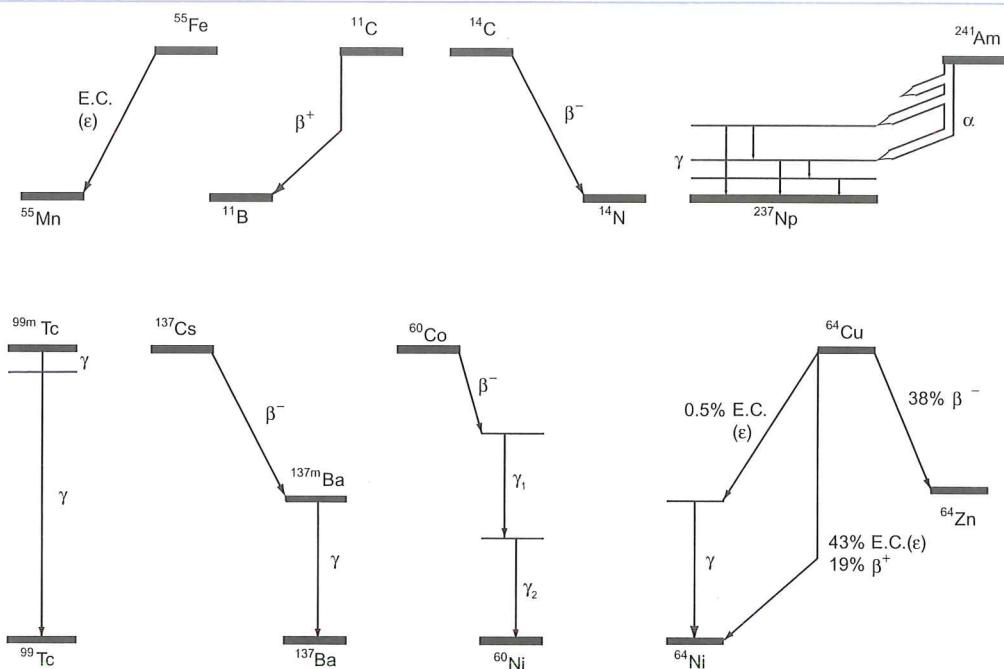


Figure 1-10

Some examples.

the parent, then during this time 100 daughter nuclei will also decay. The mixture of parent and daughter will have an activity that is two times that of the activity of the parent alone. We must be aware of this. An example of ingrowth is  $^{137}\text{Cs}$  ( $T_{1/2} = 30.17$  years), with  $^{137\text{m}}\text{Ba}$  ( $T_{1/2} = 2.55$  minutes) as daughter. An example with negligible ingrowth is  $^{99\text{m}}\text{Tc}$  ( $T_{1/2} = 6.0$  hours) with  $^{99}\text{Tc}$  ( $T_{1/2} = 2.1 \cdot 105$  years) as a daughter. In this situation, there is no ingrowth: the activity only decreases with time.

In addition, we must keep in mind that the chemical properties of the daughter are not always the same as those of the parent and that the daughter might react in its own way. This property is used in radionuclide generators. The most used is the  $^{99\text{m}}\text{Tc}$ -generator, in which  $^{99}\text{Mo}$  is adsorbed at a carrier, and the  $^{99\text{m}}\text{Tc}$  is released as  $\text{TcO}_4^-$  in physiological salt solution. By eluting the column with physiological salt, one can collect the  $^{99\text{m}}\text{Tc}$ .

### 1-6-8 Summary

The following methods of decay are of practical importance:

- $\alpha$ -decay: emission of  $\alpha$ -particles (=  $\text{He}^{2+}$ -particles =  $2\text{p} + 2\text{n}$ ).  $A$  changes by  $-4$ ,  $Z$  by  $-2$ .
- $\beta^-$ -decay: emission of  $\beta^-$ -particles (= electrons). Only  $Z$  changes by  $+1$ .
- Electron capture: only  $Z$  changes by  $-1$ . Because the hole that is formed in the electron cloud is filled immediately by an electron

from an outer shell, X-radiation is emitted. Ejection of Auger electrons also occurs.

- $\beta^+$ -decay: emission of  $\beta^+$ -particles (= anti-electrons). Only  $Z$  changes by  $-1$ . In addition, annihilation radiation develops, two photons, each with an energy of 511 keV.

From these processes, another nuclide originates. In the following decay-modes only the energy level of the nuclide changes, not its composition.

- $\gamma$ -decay: emission of photons, the composition of the nucleus remains unchanged.
- internal conversion: the energy of the nucleus is transferred to an electron in the electron cloud. This electron leaves the cloud. The vacancy that arises is filled up, whereby X-rays and Auger electrons are formed.

Combinations of these processes are possible; these may be parallel or may occur in succession.

Examples of decay are presented in decay schemes in figure 1-10. The difference between the energy level of the parent and the daughter nucleus is called the disintegration energy,  $Q$ .

The energy of the radiation is expressed in keV and MeV. An overview of frequently used and/or wellknown radionuclides is given in section 3-5-3.

## **Chapter 3**

# **Interaction of radiation with matter and shielding of radiation**

### **3-1 Introduction**

When matter is irradiated, interactions occur, which vary according to the type and energy of the radiation. These interactions are dealt with in sections 3-2, 3-3 and 3-4. In section 3-5, shielding is discussed, using the theory explained in the foregoing sections.

### **3-2 Interaction of $\alpha$ -radiation**

When  $\alpha$ -particles are formed they generally have a high energy: several MeV. As they pass through matter, the particles transfer their energy mainly to electrons, ejecting them from a shell. An electron from an outer shell fills the gap. This process eventually gives rise to a shortage of an electron in the outmost shell, and thus a positive charged atom, an ion. This interaction happens very frequently, leading to many closely consecutive ionisations. We say that the ionisation density is high. These ionisations lead to the formation of new ions. When the  $\alpha$ -particle has lost its energy it absorbs two electrons and becomes a helium atom. Therefore, there is a maximum path, which is called the *range*. Because an  $\alpha$ -particle has a large mass compared to the electrons, the path followed by an  $\alpha$ -particle is almost straight. It is very short: in air only 3 cm for  $\alpha$ -particles of 5 MeV, in solids and liquids roughly a factor 1000 less. After traversing this path, the  $\alpha$ -particle absorbs two electrons and becomes He-atom.

Because  $\alpha$ -particles have a very short range, shielding of external radiation is in fact unnecessary. Even if the source is located at only a few centimetres from the skin the  $\alpha$ -particles will not pass the epidermis. This is because the range in matter is approximately 30  $\mu\text{m}$  and for the thickness of the epidermis one can take 70  $\mu\text{m}$ . When there is internal contamination, however,  $\alpha$ -particles are extremely harmful (many ionisations over a short distance leading to damage that is difficult to repair, see Chapter 6). This is the reason that  $\alpha$ -radiation has a high radiation weighting factor (see section 5-2-4).

### 3-3 Interaction of $\beta$ -radiation

$\beta^+$ - and  $\beta^-$ -radiation are absorbed by four possible types of interactions:

- elastic collisions with electrons
- inelastic collisions with electrons
- emission of Bremsstrahlung
- emission of Čerenkov radiation.

In the case of  $\beta^+$ -radiation, there is always a fifth type of interaction:

- annihilation.

#### Elastic collisions

$\beta$ -particles are electrons. They can collide with electrons that are strongly bound in an atomic shell without ejecting this electron from its orbit; they will bounce away like a ball against a wall. The electrons close to the nucleus are strongly bound and they are important in this process. The main feature of an elastic collision is that no energy is transferred. However, the direction changes frequently. This type of interaction therefore does not lead to any loss of energy, but it causes the  $\beta$ -particles to follow a winding path.

#### Inelastic collisions

Inelastic collisions between  $\beta$ -particles and electrons are the most important ones, because they lead to the largest energy transfer from  $\beta$ -particles to matter. This interaction with electrons leads to ionisations, as discussed with  $\alpha$ -radiation. The ionisation density of  $\beta$ -radiation is several orders of magnitude smaller than the ionisation density of  $\alpha$ -radiation (roughly  $10^{-3}$ ).

Just like  $\alpha$ -particles,  $\beta$ -particles have a certain range. Because of the low ionisation density, the range is about 1000 times larger than the range of  $\alpha$ -particles with the same kinetic energy. The range as given in tables is valid for the maximum energy of the  $\beta$ -particles. This range is in fact the maximum range. For the range, there is an easy rule of thumb, see section 3-5-1.

#### Emission of Bremsstrahlung

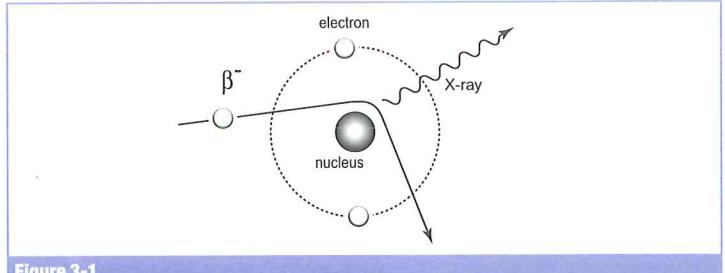
Whenever  $\beta$ -particles come very close to atomic nuclei their trajectory is deflected by the charge of the nucleus. They thereby lose energy, which is released as electromagnetic radiation (X-radiation). This radiation is called *Bremsstrahlung* (brems is the German word for brake). The higher the energy of the  $\beta$ -particles and the higher the Z-number (the charge) of the nucleus, the more important this process becomes in comparison to inelastic collisions with electrons.

The fraction ( $g$ ) of the energy converted to Bremsstrahlung equals:

$$g = 2 \cdot 10^{-4} \cdot Z \cdot E_{\beta, \max}$$

$E_{\beta, \max}$  = maximum energy of the  $\beta$ -particles in MeV  
 $Z$  = atomic number of the material.

The best materials for shielding  $\beta$ -radiation are therefore those with a low Z, for instance perspex. (see section 3-5-1).



**Figure 3-1**

Deflection of electrons near the nucleus causes Bremsstrahlung.

### Example

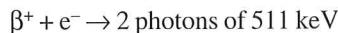
$^{32}\text{P}$  emits  $\beta^-$ -radiation with a maximum energy of 1.7 MeV. If lead (Z-number = 82) is used to shield this radiation, the fraction of energy which is converted into Bremsstrahlung is  $2 \cdot 10^{-4} \cdot 82 \cdot 1.7 = 0.028$ . In other words, approximately 3% of the energy of the particles is converted into Bremsstrahlung.

### Emission of Čerenkov radiation

The higher the energy of the particles, the greater their velocity. When energies are higher than 250 keV, the velocity of the  $\beta$ -particles approaches the velocity of light in vacuum. In a medium, however, the velocity of light is equal to  $c/n$  ( $c$  is the velocity of light in vacuum;  $n$  is the refractive index of the medium). If a high-energy  $\beta$ -particle emerges from the nucleus, its velocity may be higher than the velocity of light in the surrounding medium. In this situation the particle emits energy in the form of photons with wavelengths in the visible to violet region; this is known as *Čerenkov radiation*. When the intensity reaches a certain level, this radiation becomes visible. The amount of energy emitted is very small;  $\beta$ -particles are hardly slowed down in this type of interaction. Čerenkov radiation can nonetheless be useful because the photons can be detected.

### Annihilation

During this process the  $\beta^+$ -particle unites with an electron at the end of its path; both particles then disappear and are converted into two  $\gamma$ -photons, each of 0.511 MeV:



Thus a different kind of radiation arises, which interacts with matter in yet another way. Radiation protection in the case of against a  $\beta^+$ -emitter therefore includes shielding against the  $\gamma$ -radiation.

### 3-4 Interaction of $\gamma$ -radiation

X- and  $\gamma$ -radiation consists of photons, and photons have no charge or mass. They have no braking path, because they move with the speed of light! Unlike  $\alpha$ - and  $\beta$ -particles, which gradually change their velocity and lose energy, a  $\gamma$ -quant loses all its energy in one interaction and ceases to exist or transforms in a photon with lower energy.

The interaction of photons with matter has a certain probability, like radioactive decay. When for instance a  $\gamma$ -quant has passed a layer of lead of 1 cm and has had a 50% probability of being absorbed, then the probability of being absorbed in the next centimetre of lead will also be 50%. The formulae describing the interaction (absorption) of  $\gamma$ - and X-radiation will be the same as the formulae describing radioactive decay.

Four mechanisms are possible:

- classical scattering
- photoelectric effect
- Compton interaction
- pair production.

#### *Classical scattering*

Electromagnetic radiation can be scattered by electrons in the shells of an atom without losing energy. This form of scattering is called classical scattering, coherent scattering or Rayleigh scattering. This process takes place especially at low energies, but even then it is very rare. In X-ray diffraction equipments, this way of scattering is used to determine the structure of a crystal and of proteins.

#### *Photoelectric effect*

The photoelectric effect describes the process in which all the energy of a  $\gamma$ -photon is transferred to an electron. The electron is thereby ejected from its orbit around the nucleus and acquires a kinetic energy that is equal to the energy of the  $\gamma$ -photon minus the binding energy of the electron in the atomic shell. It is particularly the electrons that are strongly bound (electrons in the inner-most shells, like the K-shell)

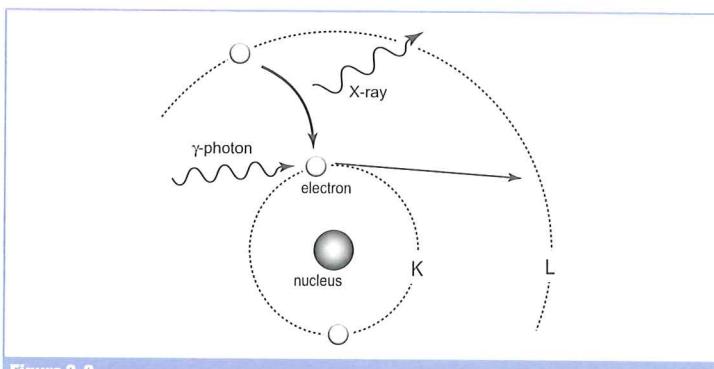


Figure 3-2

Photo-effect with a K-shell electron, followed by X-ray emission.

that participate in this process. See figure 3-2. The higher the Z-value of the nucleus, the greater is the probability of this interaction (proportional to  $Z^4$  per unit of mass).

The hole created in the shell as a result of this interaction, is immediately filled by an electron from one of the outer shells. This results in the emission of an X-ray photon.

### *Compton interaction*

Compton interaction (also called Compton scattering or Compton effect) occurs mainly with weakly bound electrons. The original  $\gamma$ -photon disappears and a  $\gamma$ -photon with a lower energy, the Compton photon, is created. The ejected electron takes with it the rest of the energy, and is called the Compton electron. The degree to which energy is distributed over the Compton photon and the Compton electron depends on the angle in which the Compton photon is emitted with respect to the original photon. See figure 3-3. With total reflection ( $\theta = 180^\circ$ ), the energy of the Compton electron is at its maximum. If the  $\gamma$ -photon only touches slightly against the electron (and does not collide), then no energy is transferred. The energy of the Compton electron ranges therefore between zero and the above mentioned maximum.

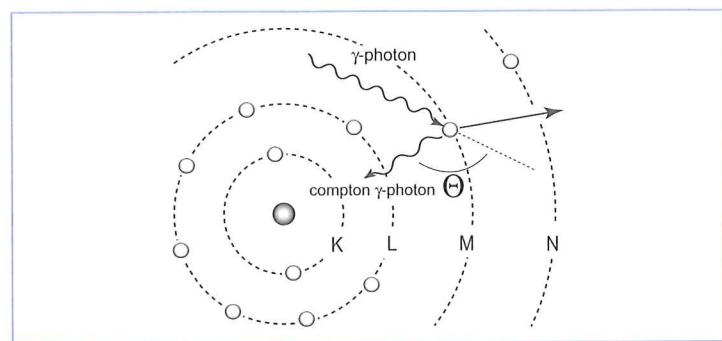


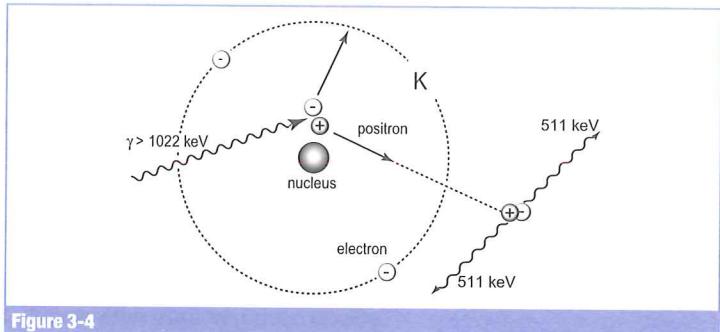
Figure 3-3

Compton effect with a weakly bound electron.

### *Pair production*

Pair production is the opposite of annihilation. Pair production occurs when a  $\gamma$ -photon transforms to an electron and a positron (identical to a  $\beta^+$ -particle). The energy of the  $\gamma$ -photon must be larger than the annihilation energy of the positron plus the electron, i.e. 1022 keV. The difference,  $E_\gamma - 1022 \text{ keV}$ , is taken up by both particles in the form of kinetic energy.

The process becomes more important as  $E_\gamma$  and the Z-value of the absorbing material increase, as the process occurs only in a strong electric field. After the positron has come to rest, the annihilation process will take place, and two  $\gamma$ -photons form, each with an energy of 511keV.

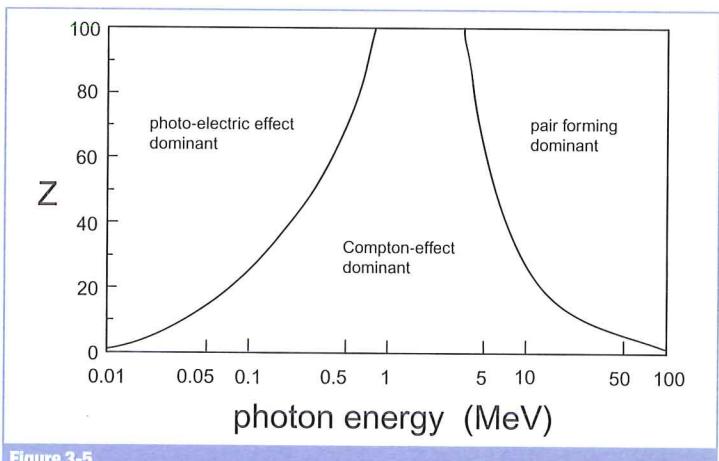


**Figure 3-4**

Pair production in the electric field of the nucleus, followed by annihilation.

### Dominance of effects

In the paragraphs above it was mentioned that the probability that the photoelectric effect or pair production will occur is dependent on the Z-value of the absorbing material and the energy of the  $\gamma$ -photon. At lower energies and high Z-values, the photoelectric effect is dominant and at higher energies (well above 1 MeV) pair production is dominant. At 2 MeV the Compton effect is dominant for all materials. See figure 3-5.



**Figure 3-5**

The dominant types of interactions as a function of the atomic number Z of the absorber and the energy of the photon radiation.

## 3-5 Shielding of radiation

### 3-5-1 Shielding of $\beta$ -radiation

$\beta$ -radiation interacts almost exclusively with electrons. How these electrons are bound is of little importance. The measure of shielding of a beam of  $\beta$ -radiation will therefore be proportional to the number of electrons in the absorption material traversed by the beam. Since the number of electrons is the same as the number of protons (and for a low Z also the number of neutrons), the measure of shielding is proportional to the amount of mass that this beam traverses.

Therefore, it is not the volume but the mass that matters. To put it another way: it is not the thickness  $d$  that matters but the mass per unit of area,  $\text{g}\cdot\text{cm}^{-2}$ . It is therefore possible to express the range of  $\beta$ -radiation in  $\text{g}\cdot\text{cm}^{-2}$ . A range expressed in this way is called the *reduced range*, and is almost independent on the nature of the specific material. For  $\beta$ -radiation with an energy of more than 150 keV, the following rule of thumb for the range is used:

$$R_\beta = 0.5 \times E_{\beta,\max} \quad \text{if } R_\beta \text{ is expressed in } \text{g}\cdot\text{cm}^{-2}, \text{ and } E \text{ in MeV.}$$

In practice, one likes to express the range in centimetres of a specific material. This range is called the linear range  $R_{\beta,\text{lin}}$ . This linear range is calculated by dividing the range by the density:



$$R_{\beta,\text{lin.}} = \frac{0.5 E_{\beta,\max}}{\rho}$$

in which  $R$  is in cm,  $E$  in MeV and  $\rho$  in  $\text{g}\cdot\text{cm}^{-3}$

By checking dimensions, one can understand that it is necessary to divide by the density. After all,  $\text{g}\cdot\text{cm}^{-2}$  divided by  $\text{g}\cdot\text{cm}^{-3}$  gives as dimension cm.

### Example

- *Given*

The maximum  $\beta$ -energy of  ${}^{32}\text{P}$  is 1.7 MeV, so the range is  $0.5 \cdot 1.7 = 0.85 \text{ g}\cdot\text{cm}^{-2}$ .

Aluminium has a density  $\rho$  of  $2.7 \text{ g}\cdot\text{cm}^{-3}$ .

The density of air is  $1.25 \cdot 10^{-3} \text{ g}\cdot\text{cm}^{-3}$ .

- *Question*

What is the linear range in aluminium and in air?

- *Answer*

The linear range in aluminium is  $0.85/2.7 = 0.32 \text{ cm}$ ; this is in agreement with the literature: 0.33 cm.

For air we expect a much larger range:

$$R = 0.85/0.00125 = 680 \text{ cm, or approximately 7 meter!}$$

We see therefore that at normal working distances air provides hardly any protection from high-energy  $\beta$ -radiation.

The range is not the only factor in the shielding of  $\beta$ -radiation. The production of Bremsstrahlung must also be taken into account, especially with high values of  $Z$ . It is therefore wise to shield  $\beta$ -radiation with materials that have a low  $Z$ , such as plastics. Sheets of plexiglas are often used for this purpose. By using layers of lead, this Bremsstrahlung is absorbed. Bremsstrahlung is emitted in all directions, including in that direction from which the  $\beta$ -radiation originates. One must be aware of this radiation if a  $\beta$ -source is not completely surrounded by sufficiently thick lead. Lead can be useful in the case of storage.

### 3-5-2 Shielding of $\gamma$ -radiation

The intensity  $I$ , the number of photons in a second, of a narrow beam of  $\gamma$ -radiation, is a function of the thickness  $d$  of the absorbing material by the same law as the decay-process:



$$I(d) = I(0) \cdot e^{-\mu \cdot d} \quad (3-1)$$

Therefore, it is possible to determine a *half-value layer*  $d_{1/2}$  (comparable to the half-life  $T_{1/2}$  in formula 1-4). This half-value layer is inversely proportional to the material constant  $\mu$ , the linear attenuation coefficient:

$$d_{1/2} = \frac{\ln 2}{\mu} = \frac{0.693}{\mu} \quad (3-2)$$

If the half-value layer is known, the linear attenuation coefficient can be calculated, and vice versa. Please note that the half-value layer and linear attenuation coefficient can also be expressed in other units. In that case  $d_{1/2}$  is expressed in  $\text{g}\cdot\text{cm}^{-2}$  and  $\mu$ , the mass attenuation coefficient, has the inverse unit  $\text{cm}^2\cdot\text{g}^{-1}$ . The linear attenuation coefficient has the unit  $\text{cm}^{-1}$ .

The relation between these two forms of  $d_{1/2}$  is

$$d_{1/2,\text{mass}} = d_{1/2,\text{lin}} \cdot \rho \quad (3-3)$$

The attenuation, in other words the shielding effect, can be calculated when either the half-value layer or the attenuation coefficient is known.

#### Example

##### - Given

The attenuation coefficient for  $\gamma$ -radiation of 1.3 MeV in lead is  $0.61 \text{ cm}^{-1}$ . Only 6.25 % (1/16-th part) of this radiation may pass through a lead shield.

##### - Question

What is the required thickness for the lead shield?

##### - Answer 1

Using formula 3-1

$$\frac{I(d)}{I(0)} = 0.0625 = e^{-0.61 \times d}$$

$$\text{So } d = -\frac{\ln 0.0625}{0.61} = 4.5 \text{ cm.}$$

##### - Answer 2

One also can follow another approach.

6.25 % equals 1/16-th part, so 4 half-value layers are necessary.

Using formula 3-2 gives

$$d_{1/2} = \frac{\ln 2}{\mu} = \frac{\ln 2}{0.61} = 1.14 \text{ cm.}$$

So  $4 \cdot 1.14 = 4.6 \text{ cm}$  lead is necessary.

Lead bricks are normally 5 cm thick; they therefore satisfy the requirement.

When you carry out a calculation, it is often wise to use both methods, as a check. A rough estimate with half-value layers will quite often give a quick and reliable result, even without a calculator. Using this half-value layer estimation one can often use the fact that for a reduction in intensity of a factor 1000 we need about 10 half-values ( $2^{10}=1024 \approx 1000$ ).

In contrast to  $\beta$ -radiation, for  $\gamma$ -radiation, particularly when the energies are not too high, the Z-value is much more important than the quantity of material per area unit. As long as the photoelectric effect is dominant (see figure 3-5) the absorption is strongly dependent on the Z-value of the absorbing material, because the chance that a photoelectric interaction will occur is proportional to  $Z^4$  per unit of mass.

At higher energies ( $> 0.5$  MeV) the Compton effect becomes dominant. Expression (3-1) still holds, but the nature of the material plays a less important role in bringing about a certain reduction in intensity. At these energies the same principle holds as with  $\beta$ -radiation: as long as the mass per  $\text{cm}^2$  remains the same, the same intensity reduction is achieved. This happens almost exactly with a  $\gamma$ -energy of 2 MeV. However, one should not conclude from the above that the choice of shielding material is unimportant at these higher energies. As a result of the Compton effect the ‘Compton photons’ develop; they can move in all directions including forwards. So especially when beams are broad, which is most frequently the case in practice, there will be forward scattering of  $\gamma$ -radiation. This forward scattered radiation has a lower energy, but adds to the transmitted radiation of the main beam; the sum leads to a higher radiation intensity than expected according to (3-1). This phenomenon is known as build-up. This *build-up* is higher in light materials than in heavy materials. In the foregoing example, this factor is about two. Using four half-value layers of concrete instead of lead, the build-up factor would have been almost five! In calculating the reduction of the radiation intensity by a shielding of thickness  $d$ , the build-up-factor  $B$  is taken into account in the following way:

$$\dot{D}(d) = B \cdot \dot{D}(0) \cdot e^{-\mu d} \quad (3-4)$$

$\dot{D}$  is the dose rate, see Chapter 5 for an explanation of the units.

In calculating shielding one must always be on the safe side and either opt for a little extra thickness or look up the build-up factor in the literature. Obviously, the radiation specialist should be consulted as well.

### 3-5-3 Overview of frequently used and/or well-known radionuclides

Table 3-1

<i>nuclide</i>	<i>T<sub>1/2</sub></i>	<i>radiation</i>	<i>E<sub>β</sub>(MeV)</i>	<i>E<sub>γ</sub>(MeV)</i>	<i>max. range (mm) in water or tissue</i>	<i>d<sub>1/2</sub> (cm) in lead</i>
<sup>3</sup> H	12.3 y	$\beta^-$	0.019		0.007	
<sup>14</sup> C	5730 y	$\beta^-$	0.156		0.3	
<sup>18</sup> F	110 min	$\beta^+, \gamma$	0.63	0.511		0.4
<sup>32</sup> P	14 d	$\beta^-$	1.7		8	
<sup>33</sup> P	25 d	$\beta^-$	0.25		0.5	
<sup>35</sup> S	88 d	$\beta^-$	0.17		0.3	
<sup>59</sup> Fe	45.1 d	$\beta^-, \gamma$	0.3 + 0.5	1.29 + 1.10		1.0 – 1.1
<sup>60</sup> Co	5.27 y	$\beta^-, \gamma$	0.3	1.17 + 1.33		1.0 – 1.1
<sup>63</sup> Ni	92 y	$\beta^-$	0.07		0.07	
<sup>66</sup> Ga	9.4 h	$\beta^+, \gamma$	4.2	0.511 + 1.04 + 2.75 + 0.83	30	0.4
<sup>67</sup> Ga	78 h	EC, $\gamma$		0.093 + 0.19 + 0.30		0.03 – 0.2
<sup>68</sup> Ga	68 min	$\beta^+, \gamma$	1.9	0.511 + 1.08	9	0.4 – 1.0
<sup>99m</sup> Tc	6.0 h	$\gamma$		0.14		0.03
<sup>123</sup> I	13.3 h	EC, $\gamma$		0.16		0.04
<sup>125</sup> I	60 d	EC, $\gamma$		0.035 + (X-ray) 0.028		0.003
<sup>131</sup> I	8 d	$\beta^-, \gamma$	0.6	0.36	2	0.3
<sup>137m</sup> Ba	2.6 min	$\gamma$		0.66		0.6
<sup>192</sup> Ir	74 d	$\beta^-, \gamma$	0.7	0.318 + 0.468	2	0.2 – 0.4

## Chapter 4

# Radiation detection

### 4-1 Introduction

Radiation detection is based on the interaction of radiation with detector material. This interaction leads to ionisations and excitations. The ionisation process was discussed earlier in Chapter 3, section 3-2. In the excitation process, an electron of one of the outer shells comes in a higher orbit, further away from the nucleus. Because the nucleus attracts the electron, energy is needed for this process, and this energy is delivered during the interaction. This *excited state* can lose its energy by emitting photons; these photons have a very low energy, compared to  $\gamma$ -radiation. They are in fact light, that is to say, in the visible spectrum; we say that that the atom or molecule scintillates.

In radiation protection, *ionisation detectors* and *scintillation detectors* are used. There are two types of ionisation detectors: gas filled and solid state (semiconductor). There are also two types of scintillation detectors: solid state scintillators and liquid scintillators. Radiation with a deep penetration potential ( $\gamma$ -radiation), is normally measured with detectors that have a high density. For radiation with a very short range ( $\alpha$ - and  $\beta$ -radiation), the detector material can have a low density: gasses are used, but also a very thin layer of solid material.

In this chapter, the detection equipment is first described with respect to the detection materials and then to its various applications in radiation protection. A summary in the form of a table is given in the last section.

### 4-2 Ionisation detectors

As the name indicates, detection is based on the formation of an ion (positive ion + electron). There are two types of ionisation detectors: the gas-filled type and the semiconductor type.

#### 4-2-1 Gas ionisation detector

This detector is a metal cylinder with a positive and a negative electrode. If a particle is captured in a detector, one or more electron-ion pairs are formed. Then, because of the electric field between the electrodes, the ion will migrate outwards towards the negative electrode (cathode) and the free electron will migrate towards the positive electrode (anode). The voltage must be high enough to prevent recombination of the electron-ion pair; this voltage must at least compensate

the mutual attractive force between the electron and the positive ion. If they would recombine to a neutral molecule, no current would be formed. When the voltage is sufficiently high, the entire charge of all the produced ions is transferred to the electrodes. The electric pulse formed this way is proportional to the number of ions formed, and thereby proportional to the energy transferred by the particle that had entered the cylinder. The *ionisation chamber* works according to this principle. This chamber is not used much in radiation protection.

By increasing the voltage, the electric pulse also increases, because the electrons in the stronger field can be accelerated to such an extent that they themselves form new ions (see figure 4-1). As long as the voltage is not too high, there is proportional amplification. In other words, the size of the pulse is still proportional to the energy transferred by the captured particle. The *proportional counter* works according to this principle.

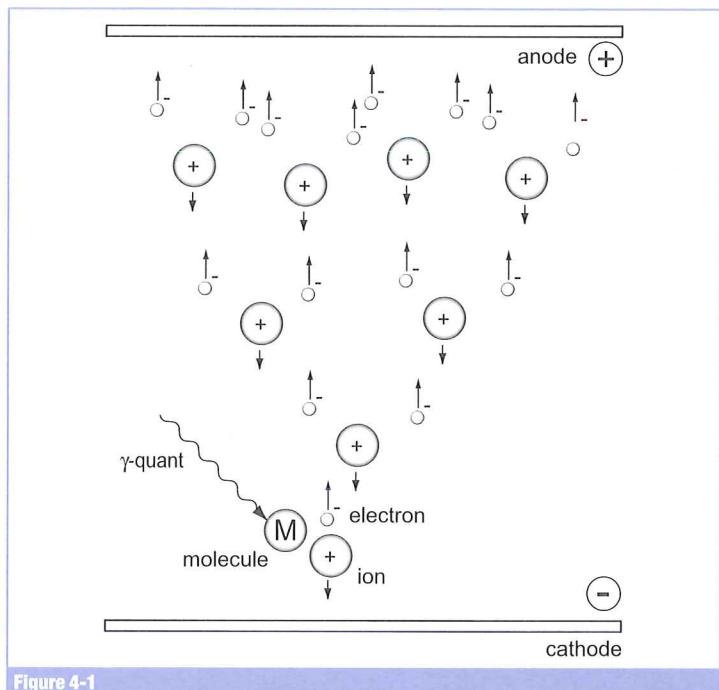


Figure 4-1

A particle ionises the molecule M, the removed electron causes new ionisations.

If the voltage is increased even more, then so many positive, slowly moving ions are formed that their total charge forms a cloud around the negative electrode (cathode) before they reach it, cancelling out the electric field. Ions move slowly because of their relatively large mass. During this shielding, there is an absence of a high voltage, and the interactions in the cylinder will not give rise to new ions or higher pulses. Therefore, with very high voltages every interaction leads, via this saturation effect, to the same pulse height, which is large and therefore easy to detect. These pulses, however, lack information about

the energy of the captured particle. This is the principle of the *Geiger-Müller counter*. Due to the saturation effect after every interaction new interactions will not be detected during a short time  $\tau$ , the ‘dead time’. At high counting rates this *dead time* may cause less counts than proportional to the activity. At high activities, the counter can almost fail to detect interactions any more!

We can correct for the dead time with the following formula

$$R_{corrected} = \frac{R_{sample}}{1 - R_{sample} \cdot \tau} - R_{background} \quad (4-1)$$

$R$  = counting rate ( $\text{cts.s}^{-1}$ )

$\tau$  = dead time per interaction (s)

In summary:

- a moderate voltage → complete collection of ions: *ionisation chamber*
- b slightly higher voltage → proportional amplification: *proportional counter*
- c high voltage → saturation: *Geiger-Müller counter* (GM-counter).

The advantage of the ionisation chamber and the proportional counter is that they provide information on the energy distribution of the radiation ( $\alpha$ - and  $\beta$ -radiation); a spectrum can be recorded. A GM-counter is very simple but gives no information on the energy spectrum. However, it is relatively cheap and very useful in radiation protection.

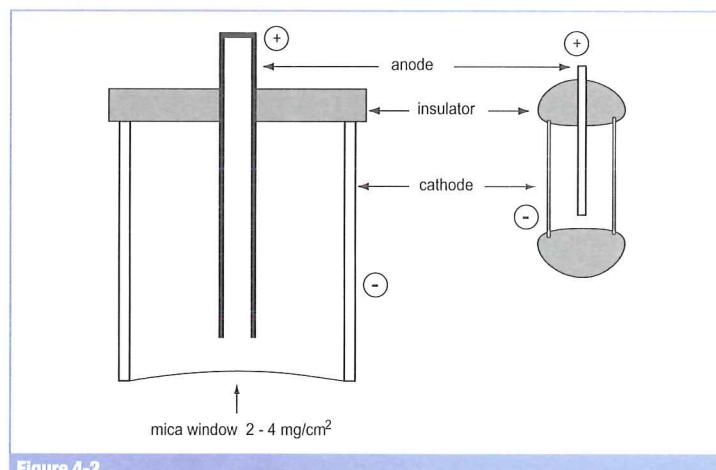


Figure 4-2

A GM-tube with a thin ‘end window’ and a small GM-tube as is used in dose rate monitors.

#### 4-2-2 Solid state detector

Electron-ion pairs as a result of interactions are also formed in solid material. If in such a material the electrodes are fixed to opposite sides and if there is a sufficient voltage difference between the two electrodes, then the electrons will move towards the positive electrode. The

ions do not move but will be neutralised by electrons from the negative electrode. For a while, there is a weak current, that is to say an electric pulse.

These solid state detectors are made of semiconductors such as germanium and silicon, to which generally a very small quantity of lithium has been added. They are called *semiconductor detectors* of the Ge-, Ge(Li)-, Si, or Si(Li)-type. If we compare the semiconductor detector with a gas ionisation detector then we see that less energy is needed to release electrons, and that the density is about 1000 times larger. This means that they are suitable for detecting  $\gamma$ -radiation, Ge(Li)-detectors in particular. Because the number of electrons formed is proportional to the energy transferred, this type of detection is very suitable for the measurements of spectra. In radiation protection, their use is somewhat limited because they are relatively expensive and sometimes have to be cooled to lower the electrical/statistical noise.

### 4-3 Scintillation detectors

#### 4-3-1 Introduction

A scintillation detector uses the phenomenon that in some materials molecules or ions emit a photon with a wavelength close to the blue-violet/ultraviolet region when they return to the ground state.

This light signal has to be converted to an electric pulse. This is done in a *photomultiplier tube*. This is a glass vacuum tube with a window, the inside of which is coated with a thin layer of Cs/Na/K or Cs/Sb. If a photon irradiates this layer, an electron is released and is thereupon attracted, accelerated and captured by a positive electrode behind the layer. When the electron strikes this electrode, several electrons are released; these are accelerated and captured by the next electrode.

Multiplication occurs yet again, and so on. When the device contains ten of these electrodes (dynodes) with a constant voltage difference, a primary electron can give rise to between one and ten million electrons. A more appropriate name for the photomultiplier tube would therefore be ‘electronmultiplier tube’.

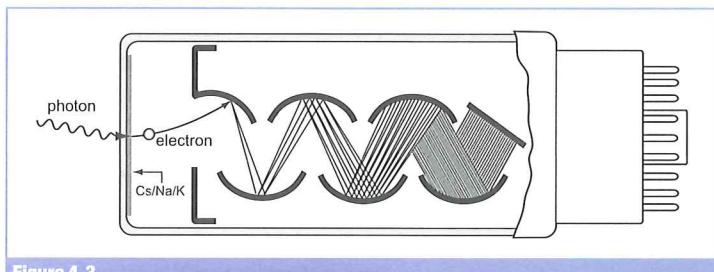


Figure 4-3

The photomultiplier tube.

In scintillation detectors, the size of the pulse is proportional to the transferred energy; this means that a spectrum can be measured. Scintillation detectors can also be divided into two types: solid state detectors and liquid detectors.

#### 4-3-2 Solid state scintillator

Solid state detectors are mainly NaI(Tl)-crystals (NaI activated with a trace of thallium). These crystals can have large dimensions, with a relatively high density, and are very suitable for measuring  $\gamma$ -radiation. Plastics such as polystyrene can be manufactured containing ‘dissolved’ scintillation molecules. A disadvantage is the low density; an advantage is that the detector can be manufactured in a variety of shapes, for instance as a flow rate counter for liquids.



**Figure 4-4**

A NaI-crystal, affixed to a photomultiplier tube.

An organic substance like *anthracene* is also used as a scintillator. Detectors made of this substance are relatively small. In contrast to NaI(Tl), this material is not hygroscopic and can therefore be used without a moisture blocking shielding. Therefore, this detector, like plastic detectors, is very suitable for detecting  $\beta$ -radiation. A classic scintillator is ZnS, often combined with a plastic. It is used in  $\alpha$ -monitors.

Apart from the above mentioned substances that scintillate almost immediately after an interaction, substances like CaF<sub>2</sub> and LiF only scintillate when they are heated. They are used in *thermoluminescence detectors*, often abbreviated to TLD. These detectors are exceptionally suitable for personal dosimetry for the following reasons:

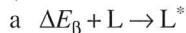
- The excited state will only give rise to scintillations during heating; at normal temperatures, the captured energy is only very slowly dissipated.
  - After these substances have been heated (the read out), they can be used again.
  - They can be used for both large and small radiation doses; the number of scintillations is proportional to the dose over a large range.
- In standard personal dosimetry, film badges (photographic detection, see section 4-4) are therefore replaced by these solid-state scintillators, most of which are lithium fluoride detectors.

**4-3-3 Liquid scintillators**

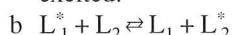
A *liquid scintillator* consists of an organic solvent containing an organic scintillator. If the substance that is to be counted is also dissolved in this solution, then maximum energy transfer occurs. In this way, the ‘weak’  $\beta$ -emitters like  $^3\text{H}$ ,  $^{14}\text{C}$ ,  $^{35}\text{S}$  and  $^{45}\text{Ca}$  can be counted very efficiently. To dissolve aqueous samples in the solution, various other substances are added. These counting liquids (counting cocktails) can be bought ready for use.

The principles underlying  $\beta$ -radiation counting are as follows:

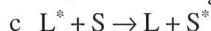
(L = solvent molecule,  $^*$  = excited state, S = scintillation molecule)



With a fraction of the  $\beta$ -energy ( $= \Delta E_\beta$ ) the solvent molecule is excited.



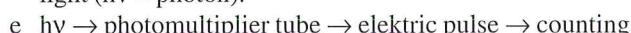
Transfer of energy between the solvent molecules.



The energy reaches a scintillator molecule.



The scintillator molecule returns to the ground state and emits light ( $h\nu = \text{photon}$ ).



Disturbances can occur in steps b) and e). If the energy transfer is disturbed by absorption of energy in molecules that do not give rise to photons when they return to their ground state (step b), then one speaks of chemical quenching. Substances such as oxygen, nitrogen compounds and organic halogen compounds can absorb energy and then emit it radiation free in the form of vibration energy. This gives the impression that less radioactivity is present. If a colouring substance (e.g. blood or urine) is present, then the photon that has been formed can be absorbed before it reaches the photo multiplier tube, in step e. This is called optical quenching. Here too, it looks as if less radioactivity is present.

When this method of counting is used, the number of photons is proportional to the energy of the  $\beta$ -particles. Because of this, only a few photons will be produced by low-energy  $\beta$ -particles, leading to a lower chance that all emitted  $\beta$ -particles are counted. In addition to this, the  $\beta$ -spectrum consists of many particles with an energy that is lower than the maximum energy, leading to less photons. These two effects are the cause that  $\beta$ -particles with low energy cannot all be counted.  $^3\text{H}$  for instance is measured in practice with an efficiency of 40%, although a maximum of 70 % can be reached.

So liquid scintillators can be used to measure the spectrum of  $\beta$ -radiation. The spectrum changes as a result of quenching (shifts towards lower pulse magnitudes). By relating the changes in the spectrum to the counting efficiency, one can correct the counts for both types of quenching; this is called quench correction.

Liquid scintillation counting and the associated quench correction are only possible if the counting sample is dissolved homogeneously or is dispersed very finely in the counting liquid, for instance in the form of a colloidal solution. If a counting sample is not soluble, e.g. a filter, then it can be helpful to extract the activity *before* the counting liquid is added. If this is not done, the measured amount of radioactivity will be much too low because of self-absorption of  $\beta$ -radiation in the filter. This especially leads to erroneous results when smear tests are done as a check on contamination!

#### 4-4 X-ray imaging

The depiction of objects and body parts began shortly after the discovery of X-rays in 1895. For this purpose, both photographic films and phosphorescent screens were used. Well-known is the image made in 1895 of the hand with ring of Mrs Röntgen, figure 4-5. It was spectacular to radiograph people and make that visible on phosphorescent screens. On funfairs one could examine the interior of persons. However, soon it was discovered that this practice is harmful to health and therefore this attraction disappeared from the scene.



**Figure 4-5**

Hand with ring of Mrs Röntgen.

The technique with films has remained current for many years. Photographic emulsions are sensitive to ionising radiation, but the detector material has very little mass (a thin layer of silver halide) and therefore the sensitivity to X-ray and  $\gamma$ -radiation is low compared to the sensitivity to  $\beta$ -radiation. This problem was solved for X-rays by pressing

intensifying screens to both sides of the films. The X-rays cause scintillations in the screens and as a result the film is exposed. Both screens and films fell into disuse by the digitalization.

On a film, it is easy to see the spatial distribution of the radiation intensity. This is used in autoradiography, where a flat object (a chromatographic paper or histological section) is pressed against an X-ray film. Here the use of  $\beta$ -emitters is required. After suitable contact time the film is developed. The emerging image indicates the precise location of the radioactivity in the section under study. If necessary, the distribution of the activity can be determined quantitatively by measuring the degrees of blackness.

When working with photographic films intensifying screens were used, as already mentioned. Nowadays phosphorescent plates are used, which have the property to convert the entrapped X-rays into excited states, which, when falling back to the ground state, emit light (scintillation). Scintillation is stimulated by a scanning laser beam. Because the spot where the laser beam falls is known, both the intensity of the resultant light and the place can be digitally recorded. This process takes place in automated readers. These plates are re-used after the readout, a big advantage compared to the films and the chemical development process.

If the readout is part of the X-ray equipment, the captured image appears within a few seconds on the screen of the X-ray worker, so that he can see immediately whether the picture is in order.

Compared to photographic film a phosphorescent plate allows large spreading in exposure. The image will not be easily over- or underexposed. That does not mean that a careful choice of tube current and voltage is not necessary for an optimal result.

If an image of the patient must be monitored during medical activities, as is done with a C-tube, then another radiation detection technique is used. The primary detector is caesium iodide with a trace of thallium ( $\text{CsI:Tl}$ ). The light generated by X-rays is registered by detectors, similar to sensors in digital cameras, and processed directly to digital images.

This detection technique is also applied in the scanners of customs. A large number of detectors are mounted in an arc and are read out continuously during the passage of trucks and trailers. Here too everything is digitally processed immediately. This technique is very sensitive and it is even possible to make details visible through thick layers of metal. Behind a 30 cm thick layer of iron a copper wire of 1 mm is still detectable.

## 4-5 Application of detection equipment in radiation protection

The choice of equipment is determined by the purposes for which it is required. These purposes come under four headings:

- source identification
- quantitative counting (determining radioactivity)
- determination of the radiation level
- registration of radioactive contamination.

### 4-5-1 Source identification

Counting can be useful to obtain information on the amount of the radioactivity. Often, however, we also want to know what nuclide is involved. Therefore we need a spectrum of the sample, or in other words, we have to determine which  $\gamma$ -energies are present in the radiation. You can find this out by using equipment where the height of the registered pulse is proportional to the transferred energy. The obtained signal is still analogue. By means of an analogue-digital-converter (ADC) this signal is converted to a number that is proportional to the size of the pulse and therefore to the energy. This number is used to add one unit at the address with that number in the memory of the multi channel analyser (MCA). Therefore an address corresponds with a narrow region of the energy spectrum. This narrow region is called a channel. These channels are represented on the monitor screen in the form of a histogram, which corresponds to the energy spectrum of the  $\gamma$ -radiation (number of pulses on the vertical axis, channel numbers on the horizontal axis), see figure 4-6.

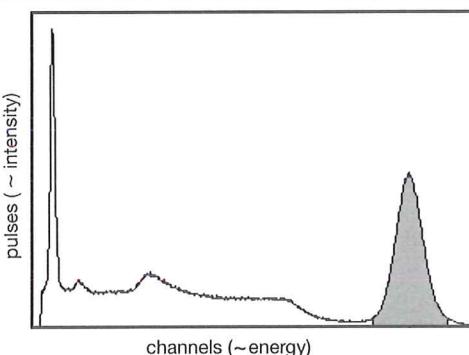


Figure 4-6

The spectrum of a  $^{137}\text{Cs}$  source, measured with a NaI(Tl)-detector. A region of interest is set around the photo peak.

If we realize that even in the smallest digital compact cameras an ADC is applied, then it is also understandable that the complete electronics, needed to make a spectrum visible, can be accommodated in the socket of the photo multiplier tube and otherwise on a plug-in chart of a PC. By choosing a region of interest with some specified channels

and summing the pulses of the channels in this region, it is possible to count this nuclide quantitatively. On the display, one sees this region of interest as a marked area of the spectrum. Usually a photo peak is chosen as region of interest. This technique to record spectra is also applied in equipment for liquid scintillators. Insight into the shape of the spectrum is necessary in order to correct for quenching. The complicated procedure for adjustment for quenching is generally carried out by the software of the liquid scintillator apparatus.

#### 4-5-2 Quantitative counting (determining activity)

Registration of interactions is the first step to determine the activity of a source. To interpret this registration it is necessary to know the efficiency of the measuring process. Among others, this efficiency is dependent on the nature of the detector, the distance from the detector to the source, the type and energy of the radiation and the emission efficiency of the radiation. (The emission efficiency is the percentage of detectable particles that is emitted at the decay process for the specific nuclide.) The ratio of counting rate to activity that combines all these factors, is called the counting efficiency,  $\epsilon$ :

$$\epsilon = \frac{R}{A} \quad (R = \text{counting rate})$$

To determine the counting efficiency it is necessary to calibrate the measuring process with a known source, a standard with equal radiation and energy.

A device that is frequently used for simple counting is the Geiger-Müller counter. All the pulses from this counter are large and of the same strength; therefore they are easily registered. Another advantage is that the high voltage does not have to be fixed at an exact level. These factors make this counter relatively simple and economical. Because a Geiger-Müller counter is filled with gas, it is best suited for  $\beta$ -radiation;  $\gamma$ -radiation will pass through the gas without giving rise to many interactions.

$\gamma$ -radiation can be counted better by means of a scintillation detector (section 4-3-2). In these detectors, the pulses have to be amplified by a photomultiplier (see section 4-3-1) and sometimes have to be analysed by a multi-channel analyser (see section 4-5-1); these monitors are therefore more expensive than the GM-counter. For very soft  $\beta$ -emitters (especially tritium), a liquid scintillation counter has to be used, as in other detection apparatuses the walls of other detectors will shield the radiation of the emitter.

#### 4-5-3 Determination of the radiation level

When we are interested in the risk of external radiation, i.e. the exposure level, then normally only X- and  $\gamma$ -radiation are relevant:  $\alpha$ - and  $\beta$ -emitters are normally completely shielded (which is possible because they have a finite range). To evaluate the exposure level of



Figure 4-7

Personal dose monitors.

X- and  $\gamma$ -radiation it is customary to use monitors with a GM-detector. Although the efficiency of these monitors for  $\gamma$ -radiation is low, they can be used for measuring exposure level, because it is only when large quantities of  $\gamma$ -photons are produced that the exposure reaches a level that should be monitored for reasons of safety.

The equipment does not differ much from the simple counting equipment mentioned in 4-5-2. The main difference is that the counter is replaced by a current meter. All incoming pulses are in fact short pulses that are averaged by the current meter. The more pulses there are, the higher the average current and the larger the deflection of the pointer. These monitors are equipped with a scale from which the exposure level can be read. Because the flow of  $\gamma$ -particles is measured, we are dealing with dose rate (see section 5-2-3). The monitors are called dose rate monitors, with the unit  $\text{gray}\cdot\text{h}^{-1}$ . Often the equivalent dose rate is given, expressed in  $\text{sievert}\cdot\text{h}^{-1}$ .



Figure 4-8

Dose rate monitor.

#### 4-5-4 Registration of radioactive contamination

Contamination monitors are used to ascertain the presence of radioactivity. These monitors do *not* measure whether a person is at risk from the radiation field caused by this activity. They measure whether there is activity that can be ingested (after contaminating the hands or the skin) or inhaled (dust particles). In general, accidentally released radioactivity will not cause a high exposure level; it can, however, be dangerous from the point of view of internal contamination. Contamination monitors are therefore much more sensitive than exposure level monitors, provided that the type of detector used is suitable.

Sometimes these contamination monitors are not calibrated for specific nuclides; in that case they give a counting rate in cps (counts per second). New types often have the option to set for a specific nuclide; they show (in digital form) the contamination in  $\text{Bq}\cdot\text{cm}^{-2}$  or in  $\text{Bq}$ .



Figure 4-9

Two types of contamination monitors. At the left a large-area-monitor with a proportional counter filled with Xe-gas (bottom and front), at the right a small GM-detector.

For a contamination with  $\beta$ -radiation (except for tritium) one uses GM-tubes with a thin window, proportional counters or scintillators like anthracene or ZnS/plastic.

The GM-tubes for  $\beta$ -radiation have a thin window so that  $\beta$ -particles with low energy can also be observed. The type of monitoring required will determine whether one opts for a detector with a large surface area (to monitor large surfaces, like tables) or for a detector with a smaller area (to monitor, for instance, glassware, glass vessels etc).



Figure 4-10

Contamination monitors with scintillation detectors. At the left a NaI(Tl)-detector for  $\gamma$ - and X-ray-radiation, at the right a ZnS-plastic detector for  $\alpha$ - and  $\beta$ -radiation.

For a contamination where *both β- and low-energy γ-radiation may be involved*, a proportional counter is often used, with a very thin, fragile window and a counting gas with a high Z-value, the noble gas xenon. Because of the thin window, they are able to detect relatively low-energy β-radiation (except for tritium), and because of the high Z they can measure the low-energy γ- and X-ray radiation. These monitors are particularly suitable for laboratories where  $^{14}\text{C}$ ,  $^{35}\text{S}$  and  $^{125}\text{I}$  are used. They are, however, rather expensive.



Figure 4-11

A check with a hand and foot monitor.

A xenon filled proportional counter is incorporated in the hand and foot monitor. Generally, this instrument has two counters for the soles of the feet, and four for the hands. Hands and feet are counted simultaneously; this is a simple and sensible check when one leaves a radio-nuclide laboratory.

For a contamination with  $\gamma$ -radiation one often can use the fact that nearly all  $\gamma$ -emitters are also  $\beta$ -emitters, so these emitters can be monitored just as effectively with GM-tubes. This is not always the case. For pure  $\beta$ -emitters like  $^{125}\text{I}$  or  $^{99\text{m}}\text{Tc}$  it is definitely preferable to use a NaI(Tl)-detector or (for low-energy  $\gamma$ -radiation) a xenon filled proportional counter. The NaI(Tl)-detectors are not large and are not particularly suitable for monitoring large surface areas; in that case the proportional counter is used.

Contamination with *tritium* cannot be measured with the normal type of contamination monitor because the very low  $\beta$ -radiation emitted by tritium cannot penetrate the window that protects the measuring chamber. Monitors that overcome this problem use a detector that has no window. Such monitors, however, are fragile, very expensive and still rather insensitive. In practice, what is generally done is to take a smear survey. A piece of (moist) filter paper is smeared over the surface one is monitoring; the paper that has swept up the material is then measured with the help of a scintillation liquid in the liquid scintillation counter. This counter can measure very sensitively low-energy emitters that have a short range. Even tritium can be measured with a reasonably high efficiency.

For *very sensitive measurements* of contamination, one should use for all  $\beta$ -emitters the liquid scintillation counter. For  $\beta$ -emitters one should use the  $\gamma$ -counter; large NaI-crystals are used to detect the radiation. In both cases smear surveys are made. By choosing a long counting time, one is able to measure very small contaminations.

#### 4-5-5 Monitoring at the gate

Radioactivity may not just be imported, exported or transported without a licence. For this reason customs check if radioactivity is present in cargo. Also companies which process scrap want to know if the scrap contains radioactivity before it enters the company area. In these cases monitoring at the gate takes place.

One wants to detect radioactivity as sensitive as possible. For this aim customs and scrap companies have very sensitive detection gateways, with large built-in scintillation detectors. These detectors may be of the NaI(Tl) type or of plastics, and have large volumes in the order of litres. Large volumes make sensitive measurement of radioactivity possible, because the larger the detector volume, the larger the chance to detect  $\gamma$ -radiation. Adding  ${}^3\text{He}$  detectors enables detection of neutron radiation.

By combining a number of detectors in the access gate, is it possible to detect radioactivity fast and sensitively, so the transports can keep on driving smoothly. If activities are traced the transport can be stopped before it enters the country or company. An image of such a detection gate is shown in figure 4-12.



Figure 4-12

A semi-trailer with container passes a detection gate with panels on both sides, containing large  $\gamma$ - and neutron detectors (NaI(Tl) and  ${}^3\text{He}$ ). (Photograph by R. de Goede, Belastingdienst, Customs Rotterdam).

#### 4-6 Counting errors and sensitivity

The measurement of radioactivity is in principle the counting of decaying radioactive atoms. Because of the random character of the radioactive decay, we will find a fluctuation in the counting results if a measurement is repeated. Based on the statistics of radioactive decay,  $\sqrt{N}$  is taken as the *standard deviation* ( $N$  is the number of collected counts). This number of counts increases with a longer counting time, as does the statistical accuracy, because the *relative standard deviation* is:

$$\frac{\text{standard deviation}}{\text{number of pulses}} = \frac{\sqrt{N}}{N} = \frac{1}{\sqrt{N}} = \frac{1}{\sqrt{R \cdot t}} \quad (R = \text{counting rate})$$

The relative standard deviation is inversely proportional to the square root of the counting time. If, for instance, the counting time is four times longer, the relative standard deviation is two times smaller.

In the counting of radioactive samples, background radiation is also registered. This background radiation originates in natural radioactive nuclides and from cosmic radiation. The measurement of samples with a low counting rate is only possible if there is a significant difference between the background radiation and the counting rate of the sample. In general, this is the case if the difference between the sample and the background measurement is larger than the standard deviation in the background measurement; the standard deviation for this difference, so in the net count, is still relatively large.

Sensitivity can be defined as the minimal detectable activity. This sensitivity depends on the relative standard deviation of the counting. So longer counting and lowering the background enhance the sensitivity!

#### Example

A counter has an average background counting rate of 36 counts per minute. We need at least to count  $\sqrt{36} = 6$  counts more in one minute to have a detectable activity. If we count 10 minutes, we need  $\sqrt{360} = 19$  counts more. This is less than 2 counts per minute. The sensitivity of the measurement is now 3 times ( $\sqrt{10}$ ) higher.

By shielding the above mentioned counter, the influence of the background radiation is reduced to 9 counts per minute. The required number of counts in 10 minutes is now  $\sqrt{90} = 9.5$  counts. This is about 1 count per minute. By decreasing the influence of the background radiation by a factor 4, the sensitivity is doubled ( $\sqrt{4}$ ).

#### Conclusion:

The sensitivity of measurements increases by lowering the background count rate. A longer counting time increases the sensitivity and lowers the standard deviation.

If a monitor has a needle indicator, we have to look at the fluctuation of the indicator during background measurement. This is the relevant factor for the sensitivity. One can only conclude that there is an activity present, if the indication is clearly larger than the normal deflection of the needle (due to the background) added up to the fluctuation.

#### Example

A contamination monitor indicates an average of 8 pulses per second above a clean surface. The indicator has a fluctuation of  $\pm 3$  cps. Thus there is a measurable contamination when the indicator shows at least 11 cps. If it is given that 1 cps equals  $0.3 \text{ Bq} \cdot \text{cm}^{-2}$ , the smallest detectable surface contamination is  $3 \cdot 0.3 = 1 \text{ Bq} \cdot \text{cm}^{-2}$ .

## 4-7 Overview of detectors used in the radiation protection

When in doubt, the radiation specialist should always be consulted about the type of monitor to be used.

*Table 4-1 Overview of detectors and their applications.*

	Measurement of $\beta$ -radiation	Measurement of $\gamma$ -radiation
<b>Ionisation detectors</b>		
GM-counter (thin window)	Contamination	Dose rate
GM-detector (thick window)	Dose Rate	Dose rate
Proportional counter (thin window, high Z)	Contamination	Contamination for low energy $\gamma$ -emitters like $^{125}\text{I}$ Dose rate
Ge(Li)	—	Identification of nuclides
<b>Scintillation detectors</b>		
NaI(Tl)	—	Contamination Identification
Anthracene/ZnS	Contamination	
TLD	Dose	Dose
Scintillation liquid	Contamination	Low-energy $\gamma$ -rays

## Chapter 5

# Quantities and units in radiation protection

### 5-1 Introduction

A quantity is used for a (physical) concept such as length, time, mass etc. A unit is an agreed measure of a quantity, such as metre, second, kilogram. In Chapter 1, the quantity activity ( $A$ ) was defined, with the unit becquerel (Bq); in this chapter we will focus on the quantities and units that provide a measure of the risk of radiation.

Note that units are written in lower case, e.g. sievert, becquerel, etc.; capitals are used for the abbreviations (Bq, J), if they are derived from personal names such as Becquerel and Joule. The definitions in this chapter are based on the Euratom Directive 96/29 and on ICRP-60, ‘1990 Recommendations of the ICRP’.

### 5-2 Definitions of quantities and units

#### 5-2-1 Introduction

Both charged particles ( $\alpha$ -particles,  $\beta$ -particles and electrons) and uncharged particles impart their energy to matter by ionisations and excitations. As ionisations cause radiation damage in biological systems, the number of ionisations is a good measure of this damage.

#### 5-2-2 Exposure

The quantity *exposure* expresses for  $\gamma$ - and X-radiation the number of ionisations in dry air. The unit for the exposure is the *röntgen* (R). A radiation field with an intensity of 1 R causes  $2.58 \cdot 10^{-4} \text{ C} \cdot \text{kg}^{-1}$  in dry air (C = coulomb, unit for electric charge). The time derivative of the exposure (the increase of exposure in time) is the exposure-rate, with the unit  $\text{R} \cdot \text{h}^{-1}$ . Because the exposure is solely defined for  $\gamma$ - and X-radiation in air, this quantity is no longer used in the Netherlands; in Anglo-Saxon literature however, the exposure is still frequently used.

#### 5-2-3 Absorbed dose

In defining a quantity for radiation damage, the property that each ionisation transfers energy is of relevance. This implies that the number of ionisations in matter is proportional to the energy that is imparted. The *absorbed dose* ( $D$ ) is defined as the energy absorbed per unit mass, i.e. as the mean energy imparted by ionising radiation to matter in a volume element divided by the mass of the matter in that volume element. The absorbed dose is often called the dose. The unit of ab-

sorbed dose is the *gray* (Gy); 1 gray equals 1 joule per kilogram ( $J \cdot kg^{-1}$ ) tissue. 1 Gy is approximately 100 R. The dose rate has as unit  $Gy \cdot h^{-1}$ .

Note that the transferred amount of energy is normally very low. The quantity energy is only chosen because it is a good measure of the number of ionisations. A dose of 1 gray, 1 joule per kilogram, will normally result in serious radiation damage, but will at the same time merely result in a temperature increase of  $0.00024^{\circ}C$ .

#### 5-2-4 Equivalent dose

The dose expressed in gray, is a measure of the transferred energy in the form of ionising radiation. A dose with high ionisation density radiation will cause more damage than an equal dose with low ionisation density. A high ionisation density is more likely to result in two damages at the opposing strands of the DNA molecule simultaneously, with a corresponding higher probability of breaking the DNA molecule (see Chapter 6). Due to its higher ionisation density,  $\alpha$ -radiation is more harmful than low ionisation density radiation such as X-rays or  $\gamma$ -rays.

In radiation protection, the absorbed dose averaged over a tissue or organ and weighted by a factor related to the type and energy of the radiation, is of interest. This is called the equivalent dose and is calculated as follows:

$$H_T = w_R \cdot D$$


The subscript  $T$  in  $H_T$  stands for the tissue (or organ),  $w_R$  is the *radiation weighting factor* (selected for the type and energy of the radiation) and  $D$  is the dose averaged over the tissue (or organ) and measured in Gy. Since  $w_R$  is dimensionless, the unit for equivalent dose is the same as for absorbed dose, namely  $J \cdot kg^{-1}$ . Its name is *sievert* (Sv). Examples of  $w_R$  values that are in current use are 1 for  $\beta$  and  $\gamma$ -radiation, and 20 for  $\alpha$ -radiation. For neutrons,  $w_R$  is a function of the energy, and varies between 5 and 20.

The increase of the equivalent dose in time is the equivalent dose-rate, with the unit  $Sv \cdot h^{-1}$ .

#### 5-2-5 Effective dose

An equivalent dose for the whole body correlates with a higher risk than an equivalent dose in a part of the body, and an equivalent dose in (for instance) the bone marrow correlates with a higher risk than the same equivalent dose in the bladder. While the equivalent dose proved to be a useful quantity for expressing doses to the whole body or individual organs, it was recognised that another quantity was needed to express partial body doses in terms of an equivalent dose to the whole body. In order to meet this need, the concept of *effective dose* was developed. It is defined as the sum of the weighted equivalent doses in all tissues and organs of the body and is symbolised by the letter  $E$ .

The unit (as for the equivalent dose) is the Sv. The effective dose expresses in just one number the risk of radiation exposure. For the calculation of the effective dose, the equivalent dose of each tissue is multiplied by an appropriate *tissue weighting factor*,  $w_T$ , for that tissue and all the products are summed.

$$E = \sum w_T \cdot H_T$$

The  $w_T$  values take into account the probability that radiation will induce a given type of cancer or genetic effect, the probability that a given cancer is lethal, the years of life that will be lost as a result of death, and the associated healthcare and psychological impacts. They all add up to unity. The following example illustrates the calculation of the effective dose.

### Example

An equivalent dose of 3 sievert in the lungs combined with a simultaneous equivalent dose of 2 sievert in the thyroid yields the following effective dose: weighting factor lungs:

0.12, so the contribution is  $H_{\text{lung}} = 0.12 \times 3 \text{ Sv} = 0.36 \text{ Sv}$   
weighting factor thyroid:

0.05, so the contribution is  $H_{\text{thyroid}} = 0.05 \times 2 \text{ Sv} = 0.10 \text{ Sv}$

The effective dose in this case is

+  
0.46 Sv

Table 5-1 The relevant quantities and units.

Quantity Name	Description	Unit Name	Description
Activity (A)	Strength of the source	becquerel (Bq)	1 Bq = 1 decay per second
Dose (D)	Damage to tissue, gray (Gy) with energy as a measure		1 Gy = 1 joule·kg <sup>-1</sup>
Equivalent dose (H) or ( $H_T$ )	Better measure of damage to tissue	sievert (Sv)	$w_R \cdot$ absorbed dose ( $w_R$ is dependent of the kind of radiation)
Effective dose (E)	Measure of risk	sievert (Sv)	Sum of $w_T \cdot H_T$ $w_T$ = tissue weighting factor, dependent of tissue T

As mentioned earlier, both the equivalent and the effective dose are expressed in sievert. This may cause confusion if language is improperly used. An example of improper use of language is: ‘a dose of 2 Sv on the thyroid’.

- Does ‘a dose’ mean ‘an absorbed dose’? Then the unit should be Gy and not Sv!
- Or does ‘a dose’ mean ‘an equivalent dose’; then  $2 \text{ Sv}$  in the thyroid equals an effective dose of  $0.05 \times 2 = 0.10 \text{ Sv}$ .
- Or does ‘a dose’ mean ‘an effective dose’, as a result of an exposure of the thyroid? In that case the equivalent dose in the thyroid is  $2 / 0.05 = 40 \text{ Sv}$ !

This example shows that a careless formulation can lead to very different interpretations.

*Table 5-2 The values of the tissue weighting factors\**.

	weighting factor ( $w_T$ )
Gonads	0.20
Bone marrow	0.12
Colon	0.12
Lungs	0.12
Stomach	0.12
Bladder	0.05
Breast	0.05
Liver	0.05
Desophagus	0.05
Thyroid	0.05
Skin	0.01
Bone-surface	0.01
Remainder	0.05**
	+ 1.00

\* The values for the tissue weighting factors will be changed due to new scientific insights. Especially the value for the gonads will be lowered from 0.2 to 0.08 (because the risk of hereditary effects is considered to be lower than previously thought) and the value for the breast will be raised from 0.05 to 0.12. These changes are not yet implemented in the Dutch regulations.

\*\* The remainder can be: adrenal glands, brains, small intestines, upper large intestines, kidneys, muscle tissue, pancreas, spleen, sweetbread and uterus.

### 5-2-6 Committed dose

After an uptake of radioactive substances in the body, tissues in the body will be irradiated. The absorbed activity will disappear eventually by radioactive decay as well as by excretion. The equivalent dose caused by this uptake is expressed in the *committed equivalent dose*  $H_{50}$ . The ‘50’ indicates that the summation is over a period of 50 years for adults. The effective dose caused by this uptake is the *committed effective dose*,  $E(50)$ . The *effective dose coefficient*  $e(50)$  represents the effective committed dose caused by an uptake of 1 Bq. A better name for the  $e(50)$  would have been the *committed effective dose coefficient*.

For the different nuclides, the values for  $e(50)$  vary more than a factor  $10^6$ . This is caused by the fact that the  $e(50)$  is dependent not only on physical decay and the rate of excretion but also on the way the uptake takes place and the distribution of the substances through the body. Some substances can enter the body through ingestion, but not via the lungs (inhalation) and some substances concentrate in specific sensitive organs, for instance iodine in the thyroid. The  $e(50)$  is also strongly dependent on the kind of radiation. The energy of  $\alpha$ -radiation is completely absorbed in the body, due to the small range of  $\alpha$ -radiation, but most of the  $\gamma$ -photons with a high energy do not interact with the body and leave the body without giving rise to ionisations. In Chapter 9, section 9-2, some values of  $e(50)$  for inhalation are given.

### 5-3 Orders of magnitude for the effective dose

The annual dose limit for occupational exposure is 20 mSv (0.02 Sv). This limit is hardly ever reached; normally the occupational dose will be less than 1 mSv. The dose for all members of the public as a result of non-occupational exposure is about 2.4 mSv in a year (see table 5-3). When a person is irradiated during cancer therapy the absorbed dose in some tissues can rise to many sieverts (spread out over time). The effective dose will be lower!

Table 5-3 The effective dose in mSv/year for the Dutch population (Health Council, 2007).

Sources	Effective dose mSv/year
cosmic radiation	0.3
external radiation (building material, emanating from soil)	0.4
$^{40}\text{K}$ in the body	0.4
radon in houses	0.8
artificial sources	0.02
X-ray diagnostics (average for the population)	0.50
occupational dose (contribution to population dose)	< 0.01
total average dose	2.4 mSv/year

### 5-4 Old and new names for quantities and units

Names and symbols have changed over time. In table 5-4, an overview is given of the changes in the *quantities*, in the 1990 Recommendations of the International Commission on Radiological Protection (ICRP 60).

Table 5-4 Old and new symbols and names for quantities used in radiation protection.

<i>Old name</i>	<i>Old symbol</i>	<i>New name</i>	<i>New symbol</i>
Dose equivalent	$H, H_T$	Equivalent dose	$H (H_T)$
Effective dose equivalent	$H_E$	Effective dose	$E$
Committed dose equivalent	$H_{50}$	Committed equivalent dose	$H_{50}$
Committed effective dose equivalent	$H_{E, 50}$	Committed effective dose	$E(50)$

Especially in the Anglo-Saxon literature, old *units* are sometimes used. The old unit for the absorbed dose is the *rad*, radiation **a**sorbed **d**ose, and the old unit for the equivalent dose is the *rem*, the **r**öntgen **e**quivalent **m**an. 100 röntgen  $\approx$ 100 rad = 1 gray, and 100 rem = 1 sievert.

# Chapter 7

# Regulations

## 7-1 Terminology

In the European Directive 96/29/EURATOM, and in the Dutch Decree Radiation Protection, an *exposed worker* is defined as a person who is subject to an exposure incurred at work that is liable to result in doses exceeding the dose limits for members of the public (i.e. an effective dose higher than 1 mSv). This means that a worker whose dose is liable not to exceed the limit of 1 mSv is by definition not an exposed worker: a worker can apply radiation without being an exposed worker. In this book the term *radiation worker* is used for both groups, i.e. for the exposed worker as well as for the worker who applies radiation without a liability to exceed a dose of 1 mSv.

In the Directive and in the Dutch Decree Radiation Protection, a human activity that can increase the exposure of individuals to radiation from an artificial source is defined as *a practice*. A human activity that can increase exposure to radiation from natural radiation sources in cases where natural radionuclides are or have been processed in view of their radioactive or fissile properties, is also called *a practice*. Work that involves just the presence of natural radiation sources and that can lead to a significant increase in the exposure of workers or members of the public which cannot be disregarded from the radiation protection point of view, is defined as *a work activity*. Many rules for ‘practices’ and ‘work activities’ are the same. In this book, ‘practices’ and ‘work activities’ are both called *applications*. The word ‘application’ has a broad meaning: it refers in this book to production, storage, processing, handling, use, and disposal of radioactive substances, and it refers to the operation and storage of equipment emitting ionising radiation.

The Directive stipulates that the application of ionising radiation is only allowed under supervision of a *qualified expert*. Depending on the responsibilities there are co-ordinating and supervising experts, and terms such as ‘local radiation specialist’, ‘responsible radiation specialist’ and ‘radiation protection officer’ are used. In this book, we use the term *radiation specialist*. Most of the time, this term will refer to a local supervising specialist, because this is the person supervising the radiation worker.

## 7-2 The system of dose limitation

The system of dose limitation is based on three factors. In order of application they are

- justification,
- optimisation,
- limitation.

The next paragraphs will clarify these factors.

In medical applications, it is not possible to convert the third step, the limitation of the acquired dose, in practicable limits for the patient. Therefore, justification and optimisation are worked out in more detail for these applications. For the justification, it holds that the doctor, irrespective of the result of the justification procedure by the government during the process of authorisation, must evaluate if an individual radiological treatment is justified. For optimisation in medical applications, the procedures are laid down in the Decree Radiation Protection.

## 7-3 Justification

Before ionising radiation is applied, it must first be ensured that the application is justified, in particular that the benefits outweigh the health detriment it may cause. If there are alternative methods, these should be considered in this procedure. Justification of the application should be based not only on weighing the own pros and cons but particularly from a point of view of the society as a whole, including future generations. This means that generation of radioactive waste is explicitly part of this weighing procedure.

For some applications the justification procedure is already carried out and formalised by the Dutch Government. For other applications the justification procedure must be carried out by the institution that wants to apply radiation, in the process of the submitting an authorisation (see paragraph 7-6-2).

So normally this justification is not a factor you will be confronted with during your daily work.

## 7-4 Optimisation, ALARA

After it has been determined that an application of ionising radiation is justified it must be ensured that the dose is kept *As Low As Reasonably Achievable*, the ALARA principle. In determining if precautions fulfil this ALARA requirement, social and economic aspects are taken into account. Different opinions in the society about social aspects can therefore be reflected as different opinions about ways to implement the ALARA requirement.

The Government and the radiation specialist have incorporated the ALARA principle in legislation and in ‘house rules’. You can keep

your dose As Low As Reasonably Achievable by adhering to the legislation and to these house rules, and in applying *the source oriented strategy*. This strategy means that controls related to the source are preferred to controls related to people. The word ‘source’ is here defined in a broad way, meaning ‘source of risk’; an X-ray apparatus is also a ‘source’.

In addition to the choice and implementation of alternative methods that confer lesser risk (mentioned under 7-3), other source related control measures include physical containment, shielding of the source or removal of the source from the workplace. People related control measures (e.g. extra instruction, wearing personal protective clothing) will be supplemented to the source related measures as deemed necessary.

In chapters on safety precautions 8 and 10, the safety precautions are treated in accordance with this source oriented strategy.

## 7-5 Dose limits

Setting limits for acquired doses is necessary to avoid too high a dose for an individual. It is possible that in the view of the society as a whole, a particular application is justified, and that the collective acquired dose is as low as reasonably achievable. However, as a result of this application, some individuals may sustain a high dose. It is for this kind of situations that the dose limit is introduced. The limits provide a safety net, only applicable after implementing the two other factors, justification and optimisation. They should not be used as a guide for a maximum permissible dose and in the practice of radiation protection they are not used that way.

The Dutch limits for the effective dose are given in the text box.

### Effective dose limits for different groups of people



Exposed workers	20 mSv/year
Apprentices/students aged between 16 and 18	6 mSv/year
Other workers	1 mSv/year
Members of the public	1 mSv/year
Foetus of an exposed worker (after reporting of the pregnancy, equivalent dose)	1 mSv

If only part of the body is irradiated, then the effective dose is calculated by multiplying the equivalent dose to that part or organ by a weighting factor:  $E = w_T \cdot H_T$ . The limit for the equivalent dose this organ is determined by dividing the limit of the effective dose by the weighting factor for the organ. For example, the weighting factor for the gonads is 0.20, so the limit is:

$$H_{\text{gonads}} = \frac{20}{0.20} = 100 \text{ mSv (a year)}$$

For hands, forearm, feet, ankles and skin (averaged over any area of 1 cm<sup>2</sup>, regardless of the area exposed) the limit for the equivalent dose for exposed workers is 500 mSv a year, to avoid deterministic effects. The limit for the lens of the eye is 150 mSv in a year.

For workers that are not exposed workers, and for members of the public most of these limits are one tenth.

The dose limits do not apply to an *emergency*. For an intervention in a radiological emergency there are – in the Netherlands – for workers and intervention personnel operational guides of 750 mSv for live saving actions and 250 mSv for saving important material goods. Exceeding these guides is allowed if workers and personnel are well informed and then only on a voluntary basis.

## 7-6 Legislation

### 7-6-1 International guidelines

The most authoritative body in the field of radiation protection is the ICRP, the International Commission on Radiological Protection. The members of this commission are experts in the field of radiation protection. The recommendations of the ICRP form the basis of most legislation for radiation protection; however, they are not legally binding.

### 7-6-2 The structure of the legislation

In Europe, the uniform basic safety standards to protect against the dangers arising from ionising radiation are laid down in Euratom Directives. Member states must ensure compliance with these standards. In the Netherlands, this is done in the Nuclear Energy Act ('Kernenergiewet') and its Decrees.

The Directive 96/29/EURATOM describes the definitions and the system of dose limitation as discussed in the previous paragraphs of this chapter.

Table 7-1 Exemption values for practices in 96/29/Euratom.

Nuclide	Activity concentration Bq. g <sup>-1</sup>	Activity Bq
<sup>3</sup> H	10 <sup>+6</sup>	10 <sup>+9</sup>
<sup>14</sup> C	10 <sup>+4</sup>	10 <sup>+7</sup>
<sup>22</sup> Na	10 <sup>+1</sup>	10 <sup>+6</sup>
<sup>32</sup> P	10 <sup>+3</sup>	10 <sup>+5</sup>
<sup>35</sup> S	10 <sup>+5</sup>	10 <sup>+8</sup>
<sup>45</sup> Ca	10 <sup>+4</sup>	10 <sup>+7</sup>
<sup>51</sup> Cr	10 <sup>+3</sup>	10 <sup>+7</sup>
<sup>99m</sup> Tc	10 <sup>+2</sup>	10 <sup>+7</sup>
<sup>125</sup> I	10 <sup>+3</sup>	10 <sup>+6</sup>
<sup>131</sup> I	10 <sup>+2</sup>	10 <sup>+6</sup>
<sup>226</sup> Ra	1	10 <sup>+4</sup>
<sup>238</sup> U	10	10 <sup>+4</sup>

It states that for practices a system of *reporting* to and prior *authorisation* by a competent authority is necessary. If the activities of radioactive substances do not exceed the *exemption values* set out in table 7-1 then the Directive states that the practices are ‘exempted’ from reporting or authorisation. These values are valid if only the mentioned nuclide is applied; if more nuclides are applied then summation rules must be used. Discharges are subject to separate limits.

For practices with electrical equipments, reporting or an authorization is normally obliged.

In the Netherlands the European Directive is implemented in the following way

No reporting is necessary if:

- the equipment has a maximum high voltage of 5 kV;
- the electrical apparatus or X-ray equipment (with a maximum high voltage of 30 kV) does not cause a dose rate exceeding 1 µSv per hour at a distance of 0.1 m from any accessible surface of the apparatus;
- an authorisation is needed.

Note that in the Netherlands reporting is also obliged if the equipment is not used, but stored for own use, installed or removed.

An authorisation is obligatory for the use of:

- equipment with a high voltage of 100 kV or higher;
- equipment for industrial radiography, processing of products, research or for medical treatment;
- equipment used for education (with some exceptions);
- an accelerator (an exception is made for electron microscopes).

The European Directives formulate measures in an unspecific way, allowing Member States to implement them in a way that is appropriate for the national situation. The way the European radiation protection measures are implemented in the Dutch legislation is discussed in the next paragraph.

### **7-6-3 The Decree Radiation Protection**

This Decree Radiation Protection implements the Euratom Directives on ionising radiation. The Decree is part of the Nuclear Energy Act. As mentioned in paragraph 7-6-1, the Euratom Directive is not only applicable to artificial sources; it also holds for natural ionising radiation. When the Decree Radiation Protection, implementing the Euratom Directive, came into force in March 2002, the number of exposed workers increased with 15 000, because cosmic radiation received by a flight crew was now also covered. However, natural sources are only covered if this radiation can be influenced. For instance  $^{40}\text{K}$ , which is naturally occurring in the human body, is excluded; radon in dwellings is also excluded.

The application of ionising radiation is only allowed under supervision of a registered radiation specialist. In the Netherlands, there are four levels, 2, 3, 4 and 5; level 2 is the highest level.

The radiation specialist must evaluate a new application and it is only allowed to start this new application after his explicit approval. For this purpose, in big institutions a system of internal licences is often used.

If there is a possibility that during the work persons can unintentionally be exposed to excessive external radiation or excessive internal contamination, then this radiation specialist must be consulted before the work has started. If there is a threat that this exposure or contamination actually will occur than the undertaking must immediately stop the work and inform, among others, the Inspectorates. The Inspectorates must also be informed if a source or an apparatus is lost, stolen or unintentionally dispersed into the environment (for instance during a fire).

Radiation workers must be given appropriate instruction about the risks, the measures to reduce these risks and the background of these measures. Special attention must be given to information of women, because of the possibility of a pregnancy.

There are two categories for exposed workers: *category A* and *category B*. Category A-workers are liable to receive a dose higher than 3/10-th of the dose limit. Category B-workers are the other exposed workers.

In the near future a further *Category C-worker* will be defined. This group includes workers who apply radiation, but with an exposure too low to be classified as 'exposed workers'.

There are two different kinds of areas. A *controlled area* is an area where it is possible to acquire an effective dose higher than 6 mSv a year. In a *supervised area*, it is possible to acquire an effective dose higher than 1 mSv a year, but less than 6 mSv a year.

A B-laboratory and a workplace in which an accelerator is placed are controlled areas; a C-laboratory and a workplace where an X-ray equipment is placed, are supervised areas.



Figure 7-1

Warning sign.



Figure 7-2

Warning sign for a controlled area.

A sign is displayed indicating the type of the area, nature of the sources and their inherent risk. This is the yellow cloverleaf (see figure 7-1), to which the text 'röntgenstraling' (= X-ray) or 'radioactieve bron' (= radioactive source) is added, and the text 'bewaakte zone' (supervised area) or 'gecontroleerde zone' (controlled area). If a controlled zone has no physical entrance control, a warning sign (see figure 7-2) should also be added with the text 'Geen toegang voor onbevoegden' (no entry for unauthorised persons).

The personal dose must be determined for an exposed worker. Usually the dose is determined by wearing a TLD badge (*TLD = Thermo Luminescence Dosimetry*). The way this regulation is formulated means that not all radiation workers are obliged to wear such a badge,

but most of them do. The registered doses are recorded in the *NDRIS*, the *Nationale Dosis Registratie- and Informatie Systeem*, until the individual has attained the age of 75 years, but in any case not less than 30 years from the termination of the work.

Category A-workers must be medically examined prior to their work, and once a year, by an approved *medical practitioner*. This examination is not necessary for category B-workers.

All information that can be relevant for the radiation safety must be recorded in the *Nuclear Energy Act File* (*Kernenergiewetdossier*). For instance, data concerning stocks, procedures, and waste management are part of the file.

The file is for record keeping. It is used by the radiation specialist to account for his actions to his management and by the management for accounting its actions to the Inspectorates.

#### **7-6-4 Other rules and regulations**

The authorization as mentioned in the Euratom Directive is called a *Kernenergiewetvergunning* (*Nuclear Energy Act Licence*). In this licence the regulations of the Decree Radiation Protection are further detailed.

In the past, the *Labour Inspectorate and the Public Health Inspectorate* have made specific Codes of Practice, but nowadays the rules in these codes are integrated in the licence. These inspectorates are charged with supervising the compliance with the provisions of the Decree and the licences.

When transporting radioactive substances one must comply to the *Decree Transport of fissionable materials, ores and radioactive substances*. In this Decree, levels are also defined below which the rules of the Decree are not applicable. Above these levels requirements are given for packaging, the maximum permissible level of radiation and contamination at the outside of the package, prescribed labels, instructions for and the training of the driver, the presence of fire extinguishers in the car, and so on. The basis for the rules is that the safety must be guaranteed in the first place by an appropriate package, so the main responsibilities are for the sender of the package. The rules for transport are very complicated. In practice, the radiation worker will always have to consult the radiation specialist for the necessary requirements.

Safety aspects that are not covered by the Nuclear Energy Act, are stated in the *Working Conditions Act* (*Arbowet*) and its Decrees. For radiation protection, the guidelines concerning calamities are relevant (for instance local assistants for calamities). The *General Administrative Law Act* (*Algemene wet bestuursrecht*) incorporates procedures to be followed by the Government when issuing a licence.



## Chapter 9

# Dosimetry in practice

### 9-1 Dose as a result of external exposure

#### 9-1-1 Introduction

External exposure refers to involuntary irradiation of (parts of) the body from a source located outside the body.

In the formulae for external exposure in this chapter, the dose  $D$  (expressed in the unit gray) is used. The conversion to the equivalent dose  $H$  (in sievert) will normally be self-evident, because the radiation weighting factor for the commonly used  $\beta$ - and  $\gamma$ -radiation is 1. However, please keep in mind that for the representation of the risk of the exposure, the effective dose  $E$ , you must multiply this equivalent dose by the relevant tissue weighting factors as given in Chapter 5, so the effective dose ( $E$ , expressed in sievert) is sometimes much lower than the equivalent dose ( $H$ , also expressed in sievert).

#### 9-1-2 The inverse square law

It is well known that the dose rate as a result of external exposure, gets larger if the source is stronger: the dose rate increases linearly with the activity ( $A$ ) of the source. Likewise, the dose rate diminishes if one is located at a larger distance from the source. However, the dose rate is not proportional to the inverse of the distance, but proportional to the inverse of the *square* of the distance,  $r$ . This ‘inverse square law’ is valid as long as the source is small (point source). For charged particles ( $\alpha$ - and  $\beta$ -radiation), this law can of course only be used within the range of these particles. Within this range, the following relationship is valid:

$$\dot{D} = C \cdot \frac{A}{r^2} \quad (9-1)$$

in which:

$\dot{D}$  = dose rate ( $\text{Gy}\cdot\text{h}^{-1}$ ) (The dot on top of the  $D$  implies a change in time)

$A$  = activity ( $\text{Bq}$ )

$r$  = distance to the source ( $\text{m}$ )

$C$  = Constant

C is, among others, a function of the type and energy of the radiation. You can calculate the dose by multiplying the exposure time by the dose rate (provided that the exposure time is short compared to the half-life).

$$D = \dot{D} \cdot t = C \cdot \frac{A \cdot t}{r^2} \quad (9-2)$$

The inverse square law is not valid very close to the source (for instance holding the source with the hands): in this situation, the source cannot be considered a point source. It is also very difficult to measure the dose rate very close to the source, because the probe of the measuring apparatus will have dimensions that are bigger than a few centimetres. If you want to make an estimate of the dose very close to the source, you can use the rules of thumb for external contamination (see formula 9-8).

### 9-1-3 Absorbtion and range for $\beta$ -radiation

Charged particles have a limited range. Formulae 9-1 and 9-2 and the rules of thumb in section 9-1-5 can only be used within this range. Outside this range, the absorbed dose is zero.

For  ${}^3\text{H}$  there will never be a dose as a result of external exposure because the range of the particles in tissue (0.007 mm) is much smaller than the insensitive epidermis of the skin (0.07 mm).

For other soft  $\beta$ -emitters ( $E_{\max} < 200$  keV:  ${}^{14}\text{C}$ ,  ${}^{35}\text{S}$ ) the range in air is lower than the usual working distance (approximately 0.3 m). At the distance of 0.3 m the dose will be zero. Closer to the source, at a few centimetres, the dose will normally be zero too, because the radiation will be absorbed in the material of the vial that contains the activity or in the fluid of the activity itself.

For a high energy  $\beta$ -emitter like  ${}^{32}\text{P}$  with  $E_{\beta,\max} = 1.7$  MeV the shielding will be taken equal to or thicker than the range of this  $\beta$ -emitter to absorb all radiation.

Ranges are given in table 3-1 and can be calculated with the formula in section 3-5-1.

### 9-1-4 Accurate calculations for $\gamma$ -emitters

For  $\gamma$ -emitters (both sealed sources and open sources) the constant in formulae 9-1 and 9-2 can be found in handbooks. This constant is called the *source constant* or gamma-radiation constant. It is represented by the symbol  $\Gamma$  and has the unit  $\text{Gy} \cdot \text{m}^2 \cdot \text{h}^{-1} \cdot \text{Bq}^{-1}$ . A selection of source constants is given in table 9-1. The formula to calculate the dose rate (for an unshielded source) is:

$$\dot{D} = \frac{\Gamma \cdot A}{r^2} \quad (9-3)$$

in which

$\dot{D}$  = dose rate ( $\text{Gy} \cdot \text{h}^{-1}$ )

$A$  = activity ( $\text{Bq}$ )

$r$  = distance to the source (m)

$\Gamma$  = source constant ( $\text{Gy} \cdot \text{m}^2 \cdot \text{Bq}^{-1} \cdot \text{h}^{-1}$ )

Table 9-1 Some source constants for  $\gamma$ -emitters.

nuclide	$\Gamma [Gy \cdot m^2 \cdot h^{-1} \cdot Bq^{-1}]$	nuclide	$\Gamma [Gy \cdot m^2 \cdot h^{-1} \cdot Bq^{-1}]$
<sup>22</sup> Na	$3.3 \cdot 10^{-13}$	<sup>111</sup> In	$8.8 \cdot 10^{-14}$
<sup>51</sup> Cr	$5.4 \cdot 10^{-15}$	<sup>123</sup> I	$4.6 \cdot 10^{-14}$
<sup>59</sup> Fe	$1.7 \cdot 10^{-13}$	<sup>125</sup> I	$3.4 \cdot 10^{-14}$
<sup>57</sup> Co	$2.3 \cdot 10^{-14}$	<sup>131</sup> I	$6.6 \cdot 10^{-14}$
<sup>60</sup> Co	$3.6 \cdot 10^{-13}$	<sup>137</sup> Cs/ <sup>137m</sup> Ba	$9.3 \cdot 10^{-14}$
<sup>67</sup> Ga	$2.5 \cdot 10^{-14}$	<sup>192</sup> Ir	$1.4 \cdot 10^{-13}$
<sup>99</sup> Mo	$2.6 \cdot 10^{-14}$	<sup>201</sup> Tl	$1.7 \cdot 10^{-14}$
<sup>99m</sup> Tc	$2.3 \cdot 10^{-14}$	<sup>241</sup> Am	$1.7 \cdot 10^{-14}$

Formula 9-3 is similar to formula 3-4, but in 3-4 the distance to the source is not included.

If we have looked up  $\Gamma$ ,  $\mu$  and the build-up factor B in literature, or use  $\Gamma$  from table 9-1 and calculate  $\mu$ , using  $d_{1/2}$  from table 3-1 (see 3-5-2), we can apply these in the complete formula for shielding:



$$\dot{D}(d) = B \frac{\Gamma \cdot A}{r^2} \cdot e^{-\mu d} \quad (9-4)$$

### 9-1-5 Rules of thumb for $\beta$ - and $\gamma$ -exposure

For  $\beta$ -emitters there are no source constants and for  $\gamma$ -emitters the source constants are not always available. In that case you can estimate the dose by using rules of thumb.

#### Rule of thumb for $\beta$ -exposure

At 30 cm of a  $\beta$ -emitter the dose rate is ( $A$  in MBq):

$$\dot{D}_\beta = 100 \cdot A \text{ } (\mu\text{Gy} \cdot \text{h}^{-1}) \quad (9-5)$$

the *mega-micro-rule* for  $\beta$ -exposure  
in which

$\dot{D}_\beta$  = dose rate ( $\mu\text{Gy} \cdot \text{h}^{-1}$ )

$A$  = activity (MBq)

This rule applies to soft tissue and for  $\beta$ -emitters between 200 keV and 2 MeV. As noted in section 9-1-3, it is only applicable within the range of the emitter.

#### Example

*Given:* During 6 minutes you divide a stock of 40 MBq <sup>32</sup>P ( $E_{\beta,\max}$  of 1.7 MeV), in smaller portions.

*Question:* What is the dose on your body (assume that the working distance is 30 cm)?

*Answer:* The range in air of <sup>32</sup>P is 8 m (see table 3-1) so the rule of thumb can be used. Using the values for the activity and the time leads to the following:  $100 \cdot 40 \cdot (6/60) = 400 \mu\text{Gy}$ .

For such a short period of time, this a considerable dose.

### Rule of thumb for $\gamma$ -exposure

At 30 cm of a  $\gamma$ -emitter the dose rate is (A in MBq):

$$\dot{D}_{\gamma} = 2 \cdot A \text{ } (\mu\text{Gy} \cdot \text{h}^{-1}) \quad (9-6)$$

the mega-micro rule for  $\gamma$ -exposure

Note that the dose caused by a gamma-emitter is about 1/50-th of the dose caused by beta-emitters with an equal activity.

At energies below 1.5 MeV the rule leads to an overestimation, which is quite large at low energies. Above 1.5 MeV the rule leads to an underestimation, if there is one  $\gamma$ -quant per disintegration. Use this rule only as a check for the calculation with formula 9-3, or if the source constant is not known.

It is possible that during decay more than one  $\gamma$ -quant is emitted. In that case, the rule of thumb can be made more sophisticated:

$$\dot{D}_{\gamma} = 2 \cdot A \cdot E_{\gamma\text{total}} \text{ } (\mu\text{Gy} \cdot \text{h}^{-1})$$

in which

$E_{\gamma\text{total}}$  = the summed energy in MeV of the emitted quanta.

#### Example

Given: 1 MBq  $^{60}\text{CO}$ .

$^{60}\text{Co}$  emits two quanta, one of 1.17 MeV and one of 1.33 MeV.

Question: The dose rate at 30 cm.

Answer: With the source constant (9-3)  $3.6 \cdot 10^{-13} \cdot 10^6 / (0.3)^2 = 4 \mu\text{Gy} \cdot \text{h}^{-1}$

With the rule of thumb (9-6)  $2 \mu\text{Gy} \cdot \text{h}^{-1}$

With the sophisticated rule of thumb  $2 \cdot (1.17 + 1.33) = 5 \mu\text{Gy} \cdot \text{h}^{-1}$

### 9-1-6 Example for X-ray equipment

The dose rate caused by X-ray equipment depends on the high voltage, the current and the distance to the tube and to the object that emits the scattered radiation. With so many variables, the calculations are difficult, and there are no easy rules of thumb. To give an idea of the order of magnitude an example is given.

#### Example

During imaging, the patient is placed in the beam so that the doctor sees a continuous image of the patient on the monitor. The dose rate in the beam at about one metre from the anode is of the order of 10 mGy per minute. A patient at one metre from the X-ray equipment and exposed for 10 minutes, receives a dose of approximately 100 mGy (the effective dose will be lower). By-standers, for example the surgeon during operations, receive only scattered radiation, about 1/10 000-th of the radiation of the direct beam, so the surgeon would receive about 0.01 mGy. In reality, because lead aprons are worn, the dose is much lower.

## 9-2 Dose as a result of internal contamination

When working with radioactive substances there is a possibility that the substances enter the body by accident. This is called *internal contamination*: the unwanted presence of radioactive substances in the body. These substances then irradiate the body or part of the body. Internal contamination can occur as a result of uptake via the mouth (swallowing, ingestion), the skin (penetration) or the lungs (breathing, inhalation).

*Contamination via the mouth* for instance occurs when you do not wear gloves, handle a contaminated object and then eat without first washing your hands. Contamination via the skin is possible because the skin is permeable to various substances.  $^{3}\text{H}$  (especially in the form of  $\text{H}_2$  or  $\text{H}_2\text{O}$ ) can even penetrate gloves.

*Contamination via the respiratory system* can occur because most manipulations with radioactive materials (mixing, pouring, pipetting, centrifugation, opening of bottles etc.) can result in aerosol formation, the formation of very small droplets of moisture or particles in the air. Aerosol formation is probably the least recognised and most underestimated source of contamination.

You can calculate the dose caused by internal contamination by using the  $e(50)$ .

$$E(50) = e(50) \cdot A. \quad (9-7)$$

where:

$E(50)$  = committed effective dose (Sv)

$A$  = activity (Bq)

$e(50)$  = effective dose coefficient ( $\text{Sv} \cdot \text{Bq}^{-1}$ )

### Example

*Given:* You perform an experiment routinely with 800 kBq of a nuclide with an  $e(50)$  of  $7.3 \cdot 10^{-9} \text{ Sv} \cdot \text{Bq}^{-1}$ . As result of not working carefully, you are internally contaminated with a 1/1000-th part of the substance each day. You work with this nuclide 200 days of the year.

*Question:* What is your committed effective dose in a year?

*Answer:* You receive 800 Bq per experiment. This is  $200 \cdot 800 = 160\,000 \text{ Bq}$  per year. A year of working carelessly results in the following dose:  $\text{Activity} \cdot e(50) = 160\,000 \cdot 7.3 \cdot 10^{-9} = 1.2 \cdot 10^{-3} \text{ Sv} (= 1.2 \text{ mSv})$ .

800 kBq is a normal amount to work with. We can therefore conclude that working carelessly during routine analyses can lead to a committed effective dose in the order of magnitude of a millisievert, and such a dose cannot be neglected. The risk of a calamity dose in the order of the limit of 20 mSv is small: you would have to swallow or inhale the complete stock!

Table 9-2  $e(50)_{inh}$  values (ICRP 68) for workers.

nuclide	$e(50)_{inh}$ [Sv/Bq]	$T_{1/2}$
$^3\text{H}$ ( $\text{H}_2\text{O}$ )	$1.8 \cdot 10^{-11}$	12.3 year
$^{14}\text{C}$ ( $\text{CO}_2$ )	$6.5 \cdot 10^{-12}$	5730 year
$^{22}\text{Na}$	$2.0 \cdot 10^{-9}$	2.6 year
$^{32}\text{P}$	$2.9 \cdot 10^{-9}$	14.3 days
$^{35}\text{S}$	$1.2 \cdot 10^{-10}$	87.5 days
$^{45}\text{Ca}$	$2.3 \cdot 10^{-9}$	163 days
$^{51}\text{Cr}$	$3.6 \cdot 10^{-11}$	27.7 days
$^{99m}\text{Tc}$	$2.9 \cdot 10^{-11}$	6.0 hours
$^{125}\text{I}$	$7.3 \cdot 10^{-9}$	60 days
$^{131}\text{I}$	$1.1 \cdot 10^{-8}$	8.0 days
$^{226}\text{Ra}$	$1.2 \cdot 10^{-5}$	1600 year
$^{238}\text{U}$	$5.7 \cdot 10^{-6}$	$4.47 \cdot 10^9$ year

The subscript ‘inh’ means that values are given for inhalation. There are separate values for ingestion, for children, for specific ways of intake etc. Inhalation is chosen because this is the most common way of contamination.

### 9-3 Dose as a result of external contamination

*External contamination* refers to contamination of the outside of the body. This external contamination results in external exposure of the body, especially the eyes and the skin. For  $^3\text{H}$ , external contamination is not relevant due to the low energy of the  $\beta$ -particles (see section 9-1-3). However, for  $\beta$ -emitters between 0.5 and 3 MeV the external exposure can be high.

#### Rule of thumb

If the contamination is 1 kBq per  $\text{cm}^2$ , with a maximum energy between 0.5 and 3 MeV then the dose rate on the skin, is

$$\dot{D} = 2 \text{ mGy} \cdot \text{h}^{-1} \quad (9-8)$$

Therefore, a contamination with a few kilobecquerels can result in a relatively high dose. The eyes are especially vulnerable as they can be rubbed without thinking and then stay contaminated for several hours. External contamination will result in external exposure, but also – owing to penetration of the skin – in internal contamination. This is a problem if the nuclides are incorporated in compounds that easily penetrate the skin.

**9-4 Examples of doses from external exposure  
and internal contamination**

If we apply the rule of thumb for  $\beta$ -radiation and the more detailed formulae for  $\gamma$ -radiation, table 9-3 is the result.

*Table 9-3 The dose caused by internal contamination and by external exposure at two distances from the source, and the time needed to lower the activity a factor 1000.*

nuclide	dose in $\mu\text{Sv}$ caused by intake of 10 kBq (internal contamination)	dose in $\mu\text{Gy} \cdot \text{h}^{-1}$ caused by 10 kBq (external exposure)	period of time (in years) for 10 kBq decay to 10 Bq	
			10 cm	100 cm
<i><math>\beta</math>-emitters</i>				
$^{3}\text{H}$ als $\text{H}_2\text{O}$	0.18	0	0	123
$^{14}\text{C}$ als $\text{CO}_2$	0.065	10	0	57300
$^{32}\text{P}$	29	10	0.1	0.4
$^{35}\text{S}$ anorg.	11	10	0	2.4
<i><math>\gamma</math>-emitters</i>				
$^{125}\text{I}$	74	0.03	0.0003	1.7

After a calculation of the dose with formulae and/or rules of thumb, it will be wise to compare the result with the values in this table. If for instance the calculation has as a result that a  $\beta$ -source of 10 MBq at 10 cm distance leads to a dose rate of  $10 \mu\text{Sv} \cdot \text{h}^{-1}$ , then the comparison with this table leads to the conclusion that you must have made a mistake in the calculation. To get a dose rate of  $10 \mu\text{Sv} \cdot \text{h}^{-1}$  at 10 cm, an order of magnitude of 10 kBq is enough.



# **Stralingsrisicoanalyse**

**Bijlagen t.b.v. Research Practicum**

**“Positron Annihilatie”**

H. Schut  
Januari 2007      1<sup>st</sup> version  
November 2013      2<sup>nd</sup> version

## Gegevens $^{22}\text{Na}$ bron gebruikt voor het RP “Positron Annihilatie”

Voor het opstellen van de Stralingsrisicoanalyse voorafgaand aan het werken met de  $^{22}\text{Na}$  bron worden hieronder de benodigde gegevens verstrekt. **De hier vermelde gegegevens vervangen die vermeld in de practicum handleiding.**

### Bron/sample configuratie:

isotoop	$^{22}\text{Na}$ (halfwaardetijd 2.60 jaar)
sterkte	4 MBq (d.d 1-6-2006)
geometrie	punt bron: bestaande uit een geringe hoeveelheid $^{22}\text{Na}$ -carbonaat ingedampt tussen twee identieke Al monsters. De Al monsters hebben een diameter van 10 mm en dikte 4 mm. Het geheel is gevatt in een perspex cylinder. (zie foto)
Bronconstante	zie tabel 9.1

### Afscherming

materiaal	Lood
dikte	5 cm

De lineike interactiecoefficient is vermeld in bijgevoegde tabel. Zie verder paragraaf 3-5-2 . Het zogenaamde “Opbouw effect” mag in de beschouwing achterwege gelaten worden (factor B = 1 in eq. 9-4)

### Werk condities:

Afstand	~ 50 cm
Tijdsduur	~ 3 uur

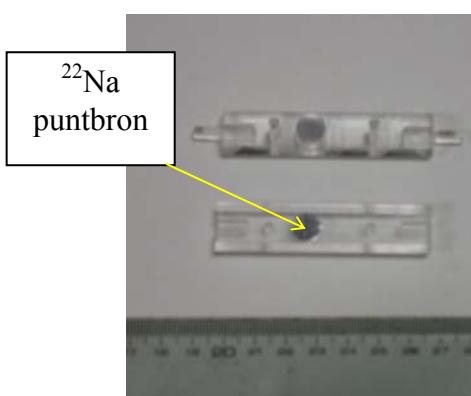
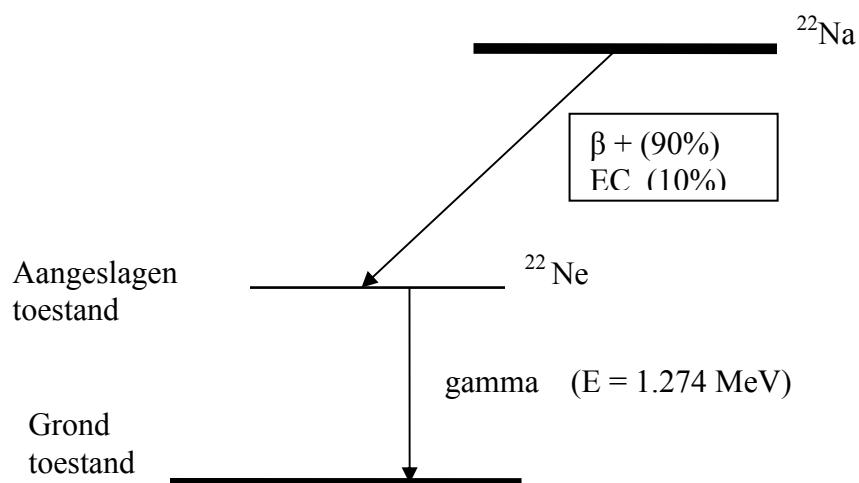


Foto van de  $^{22}\text{Na}$  bron met Al meetmonsters in opbouw. De foto toont de twee nog te verlijmen perspex cylinderhelften met daarin de twee Al monsters waarop het  $^{22}\text{Na}$  wordt aangebracht.

## Vereenvoudigd vervalschema $^{22}\text{Na}$ .

$^{22}\text{Na}$  vervalt via positron emissie (90% kans) en Electron Capture (10%) naar een aangeslagen kerntoestand van  $^{22}\text{Ne}$ . Deze toestand vervalt binnen enkele picoseconden via uitzenden een 1.274 MeV gamma naar de grondtoestand van  $^{22}\text{Ne}$ .

Het positron annihielt met een elektron in het omringende materiaal waarbij twee 511 keV gamma's worden uitgezonden (annihilatiestraling). De maximale energie van het positron bedraagt 540 keV.



Tabel A3. Totale lineaire interactiecoëfficiënten voor fotonen  
(in  $\text{cm}^{-1}$ )

Material	Density, gm/cm <sup>3</sup>	Gamma-Ray Energy, Mev											
		0.1	0.15	0.2	0.3	0.4	0.5	0.6	0.8	1.0	1.25	1.5	2
Be	1.85	.244	.220	.202	.1748	.1567	.1430	.1323	.1162	.1045	.0932	.0849	.0729
C	2.25	.335	.302	.275	.239	.2144	.1958	.1811	.1591	.1431	.1278	.1166	.0999
Na	.9712	.147	.126	.115	.099	.0886	.0809	.0748	.0657	.0590	.0530	.0482	.0415
Mg	1.741	.279	.235	.212	.185	.1643	.1497	.1384	.1217	.1092	.1975	.0891	.0770
Al	2.70	.435	.362	.324	.278	.2489	.2268	.2098	.1844	.1658	.1480	.1350	.1166
Si	2.42	.416	.336	.303	.259	.2309	.2103	.1941	.1709	.1537	.1372	.1251	.1082
P	1.83	.318	.251	.223	.190	.1698	.1548	.1427	.1254	.1129	.1008	.0919	.0798
S	2.07	.389	.298	.263	.224	.1983	.1809	.1668	.1463	.1314	.1176	.1074	.0927
K	0.87	.187	.130	.110	.092	.0816	.0741	.0684	.0599	.0538	.0480	.0439	.0381
Ca	1.55	.369	.245	.205	.169	.1496	.1356	.1254	.1097	.0983	.0877	.0803	.0699
Fe	7.86	2.704	1.438	1.065	.833	.7223	.6508	.5989	.5219	.4677	.4174	.3812	.3333
Cu	8.933	3.814	1.840	1.313	.965	.8183	.7325	.6709	.5842	.5226	.4654	.4252	.3734
Mo	9.01	9.280	3.505	2.027	1.171	.8991	.7668	.6857	.5838	.5181	.4595	.4208	.3730
Sn	7.298	11.53	4.109	2.211	1.117	.795	.6466	.5663	.4722	.4145	.3656	.3350	.2978
I	4.94	9.040	3.201	1.675	.815	.563	.4510	.3912	.3226	.2821	.2480	.2272	.2020
W	19.3	81.25	27.79	13.66	5.655	3.358	2.413	1.949	1.473	1.235	1.050	.9496	.8434
Pt	21.37	101.51	35.05	16.99	6.924	4.082	2.885	2.287	1.710	1.408	1.184	1.071	.9510
Tl	11.86	61.20	21.35	10.27	4.104	2.419	1.696	1.328	.9773	.8005	.6677	.6025	.5361
Pb	11.34	59.99	20.87	10.16	4.037	2.359	1.644	1.293	.9480	.7757	.6452	.5806	.5182
U	18.7	19.82	45.25	21.88	8.452	4.843	3.291	2.543	1.780	1.416	1.150	1.025	.9051
Nal	3.667	5.757	2.083	1.118	.568	.407	.3304	.2893	.2409	.2116	.1863	.1705	.1511
H <sub>2</sub> O	1.00	.167	.149	.118	.106	.0966	.0896	.0786	.0630	.0575	.0493	.0396	.0339
Concrete <sup>a</sup>	2.35	.397	.327	.291	.251	.2242	.2045	.1889	.1659	.1492	.1332	.1215	.1046

<sup>a</sup>Type 04.