Institut de radiophysique

RADIATION PROTECTION COURSE

Training course for the administration of unsealed radioactive substances to humans

Module II

Training course for the administration of unsealed radioactive substances to humans

Contents

CHAPTER • 1

DOSIMETRY

Course goals treated in this chapter
• *Explain the differences between absorbed dose, dose equivalent and effective dose.*
• *Apply ORaP procedures to estimate dose equivalent near a radioactive source*

1. Kerma

The ability of a photon beam to transfer its energy to electrons is characterized by its KERMA (**K**inetic **E**nergy **R**eleased per unit of **Ma**ss). In a small volume having mass Δm, the kerma K is defined by:

$$K = \frac{\Delta E_{tr}}{\Delta m} \quad \left[J \cdot kg^{-1} \right] = \left[Gy \right] \quad , \qquad (0.1)$$

where ΔE_{tr} is the sum of the initial kinetic energies of all the electrons freed by photons in the volume Δm.

The kerma unit is the joule per kilogram, which is called by the specific name gray, shortened to Gy.

In operational radiation protection, the kerma is not directly used. However, it is commonly used by testing laboratories who guarantee the coherence of other dosimetric parameters defined below.

2. Absorbed dose

Absorbed dose characterizes the quantity of energy deposited *locally* at a given location in matter. It is defined by the relationship of deposited energy ΔE per unit of mass of material Δm :

$$D = \frac{\Delta E}{\Delta m} \quad \left[J \cdot kg^{-1} \right] = \left[Gy \right] \quad . \qquad (0.2)$$

Like the kerma, the unit of absorbed dose is the gray.

Contrary to kerma, absorbed dose can be defined for any type of radiation, whether the radiation is indirectly or directly ionizing.

For a photon beam with energy typically lower than 1 MeV, the absorbed dose in the air and the air kerma are basically identical.

Absorbed dose is a purely physical parameter which does not take the biological aspects of radiation into consideration, nor the microscopic distribution of an energy deposit throughout a radiation trace.

3. Dose equivalent

When we study the effects of radiation on living organisms, we can see that the absorption of a certain amount of energy per unit of mass (absorbed dose) can have very different effects depending on the type of radiation used. In fact, the extent of effects can vary from one to two orders of magnitude depending on whether the energy was deposited by electrons or heavy charged particles. This phenomenon is caused by microscopic distribution differences of energy in matter. While gamma radiation produces a sparse microscopic distribution of ionizations (through the electrons put into movement), alpha radiation leads to a highly concentrated ionization along its trace.

The microscopic distribution of energy throughout the trace is characterized by a linear energy transfer (LET). The dose absorbed by a material during irradiation is produced by secondary electrons having a certain energy range, (i.e. a certain LET). It is thus possible, for a given radiation source to present the distribution of absorbed dose as a function of LET. Figure 1.1 presents this type of spectrum for different radiation sources. Note that the photon beams deliver the dose in a lower range of LET than heavy particles. A more detailed analysis shows that:

- For Co-60. A low LET contribution is caused by high-energy electrons produced through the Compton effect. A LET contribution higher than 10 KeV/µm is caused by low-energy secondary electrons (delta electrons).
- For 220 kV X-rays. There is no contribution from high-energy electrons.
- For 2 MeV neutrons. The spectrum extends well above 10 keV/µm This comes from protons created through elastic neutron scattering.
- For α particles. The spectrum is essentially at high LET up to about 200 KeV/µm.

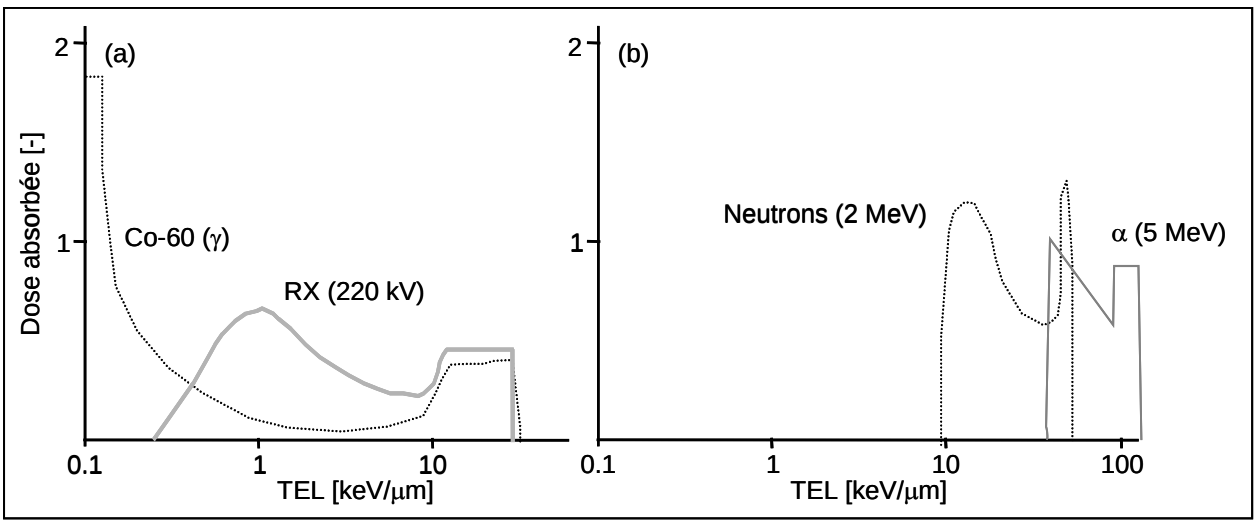


Figure 1.1: Distribution of absorbed dose as a function of LET for different types of radiation. (a) Photon sources. (b) Heavy particle sources.

3.1. Relative biological effectiveness

To consider the variation of biological effects in terms of type of radiation, radiobiology defines the notion of relative biological effectiveness (RBE). It concerns, for a given biological effect, the ratio of the absorbed dose of a reference radiation (D_{ref}) over the absorbed dose of the radiation in question (D), which is necessary to obtain the same effect level:

$$EBR = \frac{D_{réf}}{D} \quad . \qquad (0.3)$$

In radiation protection, and in order to simplify, we distinguish two types of radiation:

- Low LET radiation. This is radiation in which the main contribution of the LET spectrum is located below 10 keV/µm. This includes photonic radiation, electrons, and high-energy protons.
- High LET radiation. This is radiation in which the main contribution of the LET spectrum is located above 10 keV/µm. This includes low-energy protons, neutrons, alpha radiation and heavier particles.

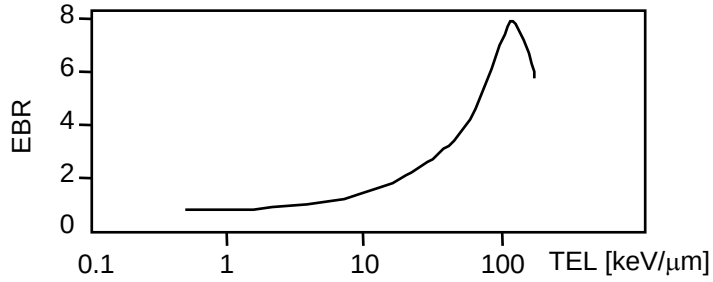


Figure 1.2: Typical look of the relative biological effectiveness (RBE) as a function of LET. We can see a net increase of RBE above 10 keV / µm.

3.2. Dose equivalent

For commonly-used radiation protection calculations, we don't use RBE which depends on tissue type and a variety of specific effects. The radiation is qualified by a *radiation weighting factor* (w_R), which does not have a unit. Table 1.1 presents the values of these factors for different radiations such as those defined in the publication ICPR 60.

Table 1.1: Radiation weighting factors w_R as defined in the publication ICPR 60.

Type of radiation		w_R
Photons of any energy		1
Electrons and muons of any energy		1
Neutrons	< 10 keV	5
	10 keV to 100 keV	10
	>100 keV to 2 MeV	20
	>2 MeV to 20 MeV	10
	> 20 MeV	5
Protons other than recoil protons	> 2 MeV	5
α particles, products of fission, heavy atoms		20

Using radiation weighting factors, we can determine a parameter derived from the absorbed dose called the dose equivalent:

$$H = w_R D_R \quad \left[J \cdot kg^{-1}\right] = \left[Sv\right] \quad . \qquad (0.4)$$

Since the radiation weighting factor is a parameter without a unit, D and H have the same units in the international system: J/kg. While these units are called grays when they describe absorbed dose, they are called *sieverts*, Sv, when they describe dose equivalent. This is used as a reminder that both biological and physical principles were taken into consideration.

If the radiation involves different radiations R of radiation weighting factors w_R inducing each to an absorbed dose D_R, the dose equivalent can be expressed through the weighted sum over the radiations R of the absorbed doses:

$$H = \sum_R w_R D_R \quad . \qquad (0.5)$$

Note also that the dose equivalent defined for radiation protection is only used to describe small values for which only stochastic effects (see further on) may appear. We will see in a later chapter that this limit is fixed at 0.5 Sv. Beyond 0.5 Sv, deterministic effects appear and the quantity of the radiation received by the organism is characterized by the absorbed dose.

3.3. Operational parameters

In current radiation protection practices, it is important to qualify a field of radiation in terms of how much danger it represents for a human being. Given that any real-life situation will be highly complex, necessary simplifications are made. In this way, a human is represented by a sphere and the dose equivalent is determined for two depths: at 0.07 mm deep for skin and at 10 mm deep for internal organs.

3.3.1. ICRU sphere

The ICRU sphere is by definition a 30 cm diameter sphere with a density equal to 1 g/cm^3 and the following composition: 76.2% oxygen, 11.1% carbon, 10.1% hydrogen and 2.6% nitrogen. The composition of this sphere is approximately the same as soft tissue. The dimensions used are roughly equivalent to the human body.

3.3.2. Ambient dosimetry

Ambient dose equivalent: H(10)*

Ambient dose equivalent H*(10) in a point P in a given field is defined as the dose equivalent at 10 mm deep into the ICRU sphere. H*(10) is considered a good approximation of the dose equivalent received by a deep organ. H*(10) is non-negligible when the radiation is relatively penetrating.

Directional dose equivalent: H'(0.07)

For less penetrating radiation and for an organ in direct contact with the radiation field (skin), we have defined *directional dose equivalent* H'(0.07). This is, in a point of a radiation field, the dose equivalent which would be given at 0.07 mm deep into the ICRU sphere.

3.3.3. Personal dosimetry

Personal deep dose equivalent: Hp(10)

The personal deep dose equivalent $H_p(10)$, using the abbreviation H_p is the dose equivalent in soft tissue beneath a thickness of 10 mm at the chest.

Personal surface dose equivalent: Hp(0.07)

The personal surface dose equivalent $H_p(0,07)$, using the abbreviation H_s is the dose equivalent in soft tissue beneath a thickness of 0.07 mm near the thorax.

Hp(10) and Hp(0.07) are the two parameters used for measuring personal dosimetry.

3.3.4. Dose equivalent near a radioactive source.

By hypothesizing that the radioactive source is discontinuous and that there is no attenuation between the source and the point of interest, it is possible to estimate the dose equivalent.

The ambient dose equivalent is defined by:

$$H^*(10) = \dot{h}(10)\frac{A\,t}{r^2} \quad , \qquad (0.6)$$

where A is the activity of the source, r is the distance between the source and the point of interest (in meters), and **h(10)** is the rate index of ambient dose equivalent. Again, the units of the constant define the units of the other variables.

In a similar way, we define the directional dose equivalent as:

$$H'(0.07) = \dot{h}'(0.07)\frac{A\,t}{(10\,r)^2} \quad , \qquad (0.7)$$

where $\dot{h}'(0.07)$ is the rate index of the directional dose equivalent at 0.1 m distance from the source. The reason the index is given at a 0.1 m distance is because this parameter is essentially used for slightly penetrating radiation with a non-negligible absorption into the air. The value of the index is thus given at 10 cm to be conservative and to overestimate any dose at a distance greater than 10 cm.

It is also interesting to know the directional dose equivalent when in direct contact with a radiation source. In this case, the application of the previous equation is obviously not possible, and we characterize the source by its surface activity A_S (activity per surface unit). The directional dose equivalent is expressed by:

$$H'(0.07) = \dot{h}_C(0.07)\,A_S t \quad , \qquad (0.8)$$

where $\dot{h}_C(0.07)$ is the index of surface directional dose equivalent.

The indexes of dose equivalent can be found in the Ordinance on Radiation Protection (ORaP). An example is given in Figure 1.3.

Radioprotection - O

Nucléide	Période	Type de désintégration/ de rayonnement	e_{inh} Sv/Bq	e_{ing} Sv/Bq	Grandeurs d'appréciation h_{10} (mSv/h)/GBq à 1 m de distance	$h_{0,07}$ (mSv/h)/GBq à 10 cm de distance	$h_{c0,07}$ (mSv/h)/(kBq/cm²)	Limite d'exemption **LE** Bq/kg ou LE_{abs} Bq
1	2	3	4	5	6	7	8	9
Co-57	270.9 d	ε, γ	6.0 E-10	2.1 E-10	0.021	100	0.1	5 E+04
Co-58	70.80 d	ε, β^+, γ	1.7 E-09	7.4 E-10	0.147	300	0.3	1 E+04
Co-58m	9.15 h	γ	1.7 E-11	2.4 E-11	<0.001	10	<0.1	4 E+05
Co-60	5.271 a	β^-, γ	1.7 E-08	3.4 E-09	0.366	1000	1.1	1 E+03
Co-60m	10.47 m	β^-, γ	1.2 E-12	1.7 E-12	0.001	20	<0.1	6 E+06
Co-61	1.65 h	β^-, γ	7.5 E-11	7.4 E-11	0.017	1000	1.6	1 E+05
Co-62m	13.91 m	β^-, γ	3.7 E-11	4.7 E-11	0.436	1000	1.8	2 E+05
Ni-56	6.10 d	ε, γ	9.6 E-10	8.6 E-10	0.260	60	0.1	1 E+04
Ni-57	36.08 h	ε, β^+, γ	7.6 E-10	8.7 E-10	0.278	700	0.8	1 E+04

Figure 1.3: Example of Annex 3 of the Ordinance on Radiation Protection (ORaP). The dose index from Column 6 corresponds to the equation (5.6), the index from Column 7 to the equation (5.7) and the index from Column 8 to the equation (5.8). Columns 4 and 5 correspond to the equations (5.11).

4. Effective dose

4.1. Effective dose

In order to compare heterogeneous irradiation situations which lead to different distributions of dose equivalent throughout the body, the ICRP has introduced the notion of *effective dose* E as the weighted sum of dose equivalent H_T to the irradiated organs and tissues T.

$$E = \sum_T w_T \cdot H_T \quad , \qquad (0.9)$$

where w_T is the radiation weighting factor for the organ or tissue T which expresses the fraction of radiation risk associated with the organ or tissue in the case where all the organs or tissues receive the same dose. Like dose equivalent, the unit of effective dose is the sievert (Sv).

Given how we define factors w_T, this parameter can also be seen as the dose equivalent received by a uniformly irradiated person for an identical radiological risk.

Table 1.2 presents the factors w_T by organ group. w_T values are values established for a reference population having an equal number of sexes and a wide age distribution. This makes them applicable to workers, the entire population and both sexes.

Table 1.2: List of w_T factors per organ group (source: Swiss Federal Ordinance on Radiation Protection).

Organ or tissue	w_T	$\sum_T w_T$
Surface of the bone, skin	0.01	0.02
Bladder, breast, liver, esophagus, thyroid, remainder[1]	0.05	0.30
Bone marrow, colon, lung, stomach	0.12	0.48
Gonads	0.20	0.20
Total		1.00

4.1.1. Intake

When intake occurs (by ingestion or inhalation), the dose is deposited continuously while the radionuclide is present in the organism. The parameter describing irradiation through intake is the committed effective dose (E50) which corresponds to the sum of the effective doses per unit of time E(t) received over the fifty years following the intake:

$$E_{50} = \int_{t_o}^{t_o + 50\ ans} E(t)\,dt \quad , \qquad (0.10)$$

where t_o is the moment of intake. In this way we can count the total dose incurred by an annual intake, even if the radiation is deposited at a later date.

[1] For this calculation, the remainder is composed of the following organs and tissues: adrenal glands, the brain, the upper part of the intestine, the small intestine, the kidneys, the muscle, the pancrease, the spleen, the thymus and the uterus.
In an exceptional situation where one of the organs or tissues making up the remainder receives a dose equivalent exceeding the highest dose in the twelve organs or tissue for which the w_T is specified, then a w_T of 0.025 must be applied to this organ or tissue and a w_T of 0.025 to the average dose in the rest of the other organs or tissue making up the remainder.

The value E(t) depends on the physical half-life of the radionuclide and certain physiological mechanisms. A model of this temporal dependence and a calculation of the corresponding dose can be found in Annex 3 of ORaP (see Figure 1.3 for an example). Thus, for a given nuclide, E_{50} can be simply estimated from the inhaled (A_{inh}) or ingested (A_{ing}) activity:

$$\begin{aligned} E_{50} &= e_{inh} \cdot A_{inh} \\ E_{50} &= e_{ing} \cdot A_{ing} \quad . \end{aligned} \qquad (0.11)$$

4.2. Link between operational parameters and effective dose

Operational parameters are those measured in practice. For an initial approximation of an external irradiation, effective dose can be estimated using the following:

$$E_{ext} = H_p(10) \quad \text{ou} \quad H^*(10) \quad . \qquad (0.12)$$

In a similar way, the dose to the skin can be estimated by:

$$H_{peau} = H_p(0.07) \quad \text{ou} \quad H'(0.07) \quad . \qquad (0.13)$$

In an intake situation, effective dose is simply estimated by:

$$E_{inc} = E_{50} \quad . \qquad (0.14)$$

Total effective dose can be obtained by summing the contributions from the external irradiation and the intake:

$$E = E_{ext} + E_{inc} \quad . \qquad (0.15)$$

Summary:

- *The kerma (K) is the energy transferred per unit of mass from non-charged particles to charged particles in the form of kinetic energy. The unit involved is the gray (Gy)*
- *Absorbed dose (D) is the energy deposited per unit of mass. The unit is the gray (Gy)*
- *Radiation which produces high-density ionizations (high LET) are, at an equal absorbed dose, more efficient in terms of biological effects compared to low-density ionizing radiation (low LET). This is caused by the fact that the mechanisms of cellular repair function much better when the extent of damage is limited.*
  - *Low LET: photons, electrons, high-energy protons.*
  - *High LET: neutrons, alpha particles, protons.*
- *The relative biological effectiveness (RBE) of a given LET radiation is the ratio, for a given biological effect, of the absorbed dose delivered by a reference radiation over the absorbed dose delivered by a given LET radiation, needed to obtain the same level of effect.*
- *The radiation weighting factor (w_R) is an approximation of RBE used in radioprotection. The values are the following:*
  - *Photons: 1*
  - *Electrons: 1*
  - *High-energy protons: 1*
  - *Protons: 5*
  - *Neutrons: 5 - 20*
  - *Alpha particles: 20*
- *Dose equivalent (H) is a simplified way to take into consideration the effect of radiation on biological tissue. Dose equivalent is defined as $H = w_R D$. The unit of dose equivalent is the sievert (1 Sv).*
- *Ambient dose equivalent H*(10) is representative of the dose equivalent received by an internal organ.*
- *Directional dose equivalent H*(0.07) is representative of the dose equivalent received by the skin.*
- *Effective dose is the dose equivalent received by a person uniformly irradiated for an identical radiological risk.*
- *Committed effective dose (E_{50}) corresponds to the sum of effective doses received over the 50 years which follow an intake.*

5. Practice questions

1. Calculate the increase in water temperature after an irradiation with an absorbed dose of 2 Gy.

2. Estimate the number of ionizations produced when a 1 MeV proton passes through a cell (approximately 1 µm in diameter). We assume that an ionization requires 30 eV.

3. Same as the previous question, but for a 1 MeV electron.

4. An irradiation of P-32 occasions an absorbed dose of 2 Gy. What is the dose equivalent?

5. Calculate the dose equivalent produced by a simultaneous alpha and beta irradiation at the following absorbed doses: D_{α}=1.4 mGy and D_{β}=10.1 mGy.

6. Calculate the ambient dose equivalent rate and the directional dose equivalent rate at 7 m distance from a Co-60 source of 500 MBq.

7. Calculate the skin dose received on the hand of a person leaning against a P-32 source with a surface activity of 1 MBq/cm^2 for one minute.

CHAPTER • 2

MEASURING RADIATION

Course goals treated in this chapter
• *Cite and describe the principles of radiation detection.*
• *Describe how to measure using liquid scintillation.*
• *Be able to list the different measurement parameters (dose, dose rate, surface contamination, sample activity, …) and explain to which situations they apply.*
• *Know how to explain the main characteristics of the different instruments: sensitivity, background noise and detection limit.*
• *Understand the significance of the response curve in energy.*
• *Be able to select the appropriate instrument for measuring a given parameter, for radiation characteristics (α, β, γ, energy) and its intensity.*
• *Be able to conduct a measurement with a measuring instrument for ambient radiation and a surface contamination monitor.*

1. Introduction

Radiation detection is based on measuring the effects generated through the interaction of a given radiation with the sensitive volume of the instrument. These effects result from the radiation-matter interaction phenomena presented in the following chapters. Radiation interactions in the sensitive volume of a detector depend on both the characteristics of the detector (composition, dimensions, etc) and on the characteristics of the radiation (α, β, γ, energy). In particular, the considerations concerning the penetration of radiation in matter seen in the previous chapters are important in order to understand the possibilities and limits of various measuring instruments. Which is why selecting the appropriate instrument for the metrology of ionizing radiation involves a preliminary evaluation. This initial consideration evaluates the specific conditions of the measurement and the following points:

- parameter to measure: absorbed dose, activity, dose equivalent,…
- type of radiation: α, β, γ, neutrons, …
- radiation energy.
- type of measurement: geometry, instant data, individual measurement…
- other considerations: difficulties, length of time …

The principles underlying the detection of ionizing radiation are extremely varied. They include measuring ionization, heat increase, luminescence and oxidation-reduction phenomena. Figure 6.1 presents an overview of the main physical principles which are used for measuring ionizing radiation.

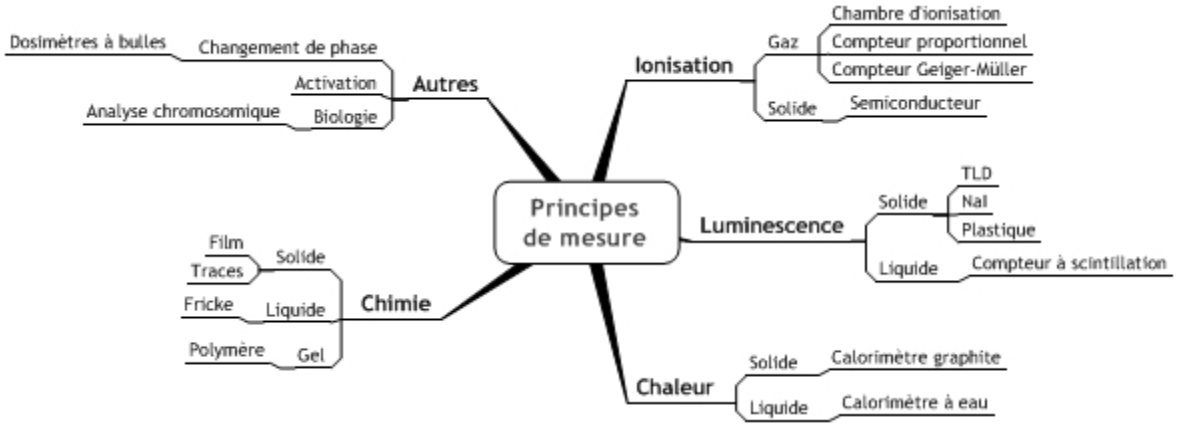


Figure 2.1: Different detection principles and their related instrument categories.

2. Ionization-based detectors

2.1. How they work

An ionizing radiation penetrating a given matter produces ionizations in the form of negative (electrons) or positive (associated ions) charges. When these ionizations occur within a gas delimited by two electrodes between which a difference of voltage is applied (see Figure 2.2) we can see a migration of positive charges toward the cathode and negative charges toward the anode[2]. In a closed circuit, an electrometer indicates the electrical current passing through the volume of gas.

The response of a gas detector depends on the voltage applied to the two electrodes. Generally, the higher the voltage, the greater the number of charges will be collected by the electrometer. Figure 2.3 presents the typical evolution of the signal of this kind of instrument in terms of the voltage applied. This characteristic curve highlights distinct zones defining the working fields in an *ionization chamber* (B), a *proportional counter* (C) and a *Geiger-Müller counter* (E).

By observing zone A in Figure 2.3, we can see that at low-voltage the signal amplitude (i.e. the number of ions collected) increases with the voltage. This is so because at a very low voltage, the electric field existing between the two electrodes is too weak to stop any recombining between negative and positive ions. As the voltage increases, the rate of recombination decreases and the charge collected by the electrometer increases.

[2] The cathode is the negative electrode and the anode is the positive electrode.

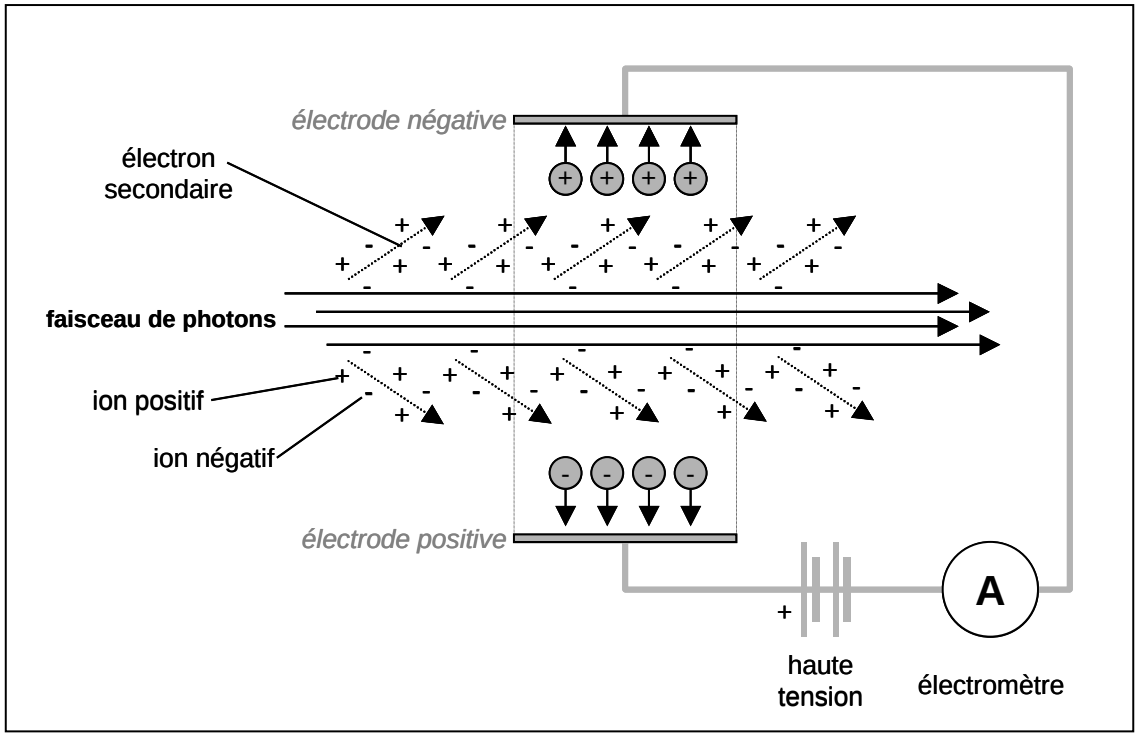


Figure 2.2: How an ionizing detector works.

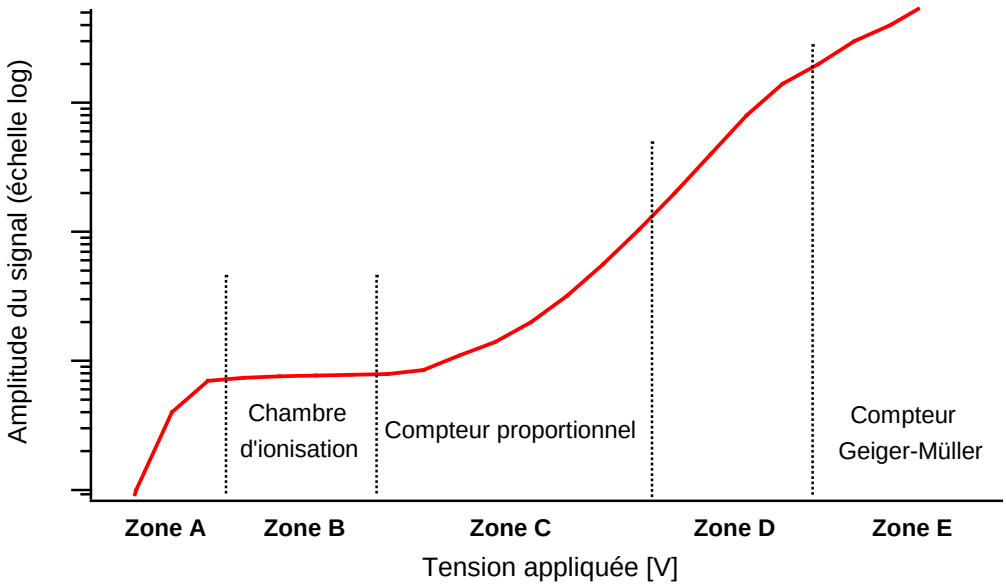


Figure 2.3: Variation of the ionization current with applied voltage.

2.2. Ionization chamber

After a certain voltage level, the detector leaves zone A and enters zone B in Figure 2.3. In this situation, an increase in voltage does not translate into a signal increase. This is because the majority of the ions created by the incident radiation reach the detector's electrodes (see Figure 2.4). The recombination rate is weak and is a function not only of the applied voltage, but also of the geometric characteristics of the chamber as well as the measured dose rate. Sensitivity is directly proportional to the volume and is also dependent on the energy of the radiation. An instrument functioning in this voltage range is called an *ionization chamber* (see the example in Figure 2.5).

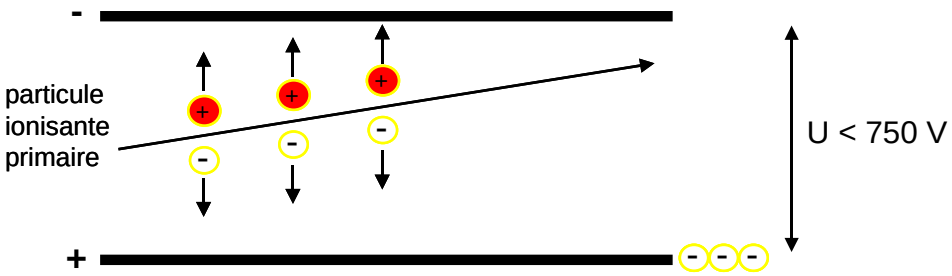


Figure 2.4: Functioning in ionization chamber mode. The primary charges produced are collected at the terminals of the electrodes. There are very few recombinations.

Figure 2.5: Example of an ionization chamber used in radiodiagnostics (Radcal chamber with a volume equal to 6 cm^3).

An important property of the ionization chamber is the material making up its wall. This is linked to the fact that, when detecting indirect ionizing radiation (like photons), the secondary charged particles produced in the wall (mainly electrons) are measured in the volume of the chamber (see Figure 2.6). The thickness and the type of material must be selected to match the specific conditions of the measurement. In the field of medical radiophysics, ionization chambers are mainly used in the dosimetry of radiotherapy beams.

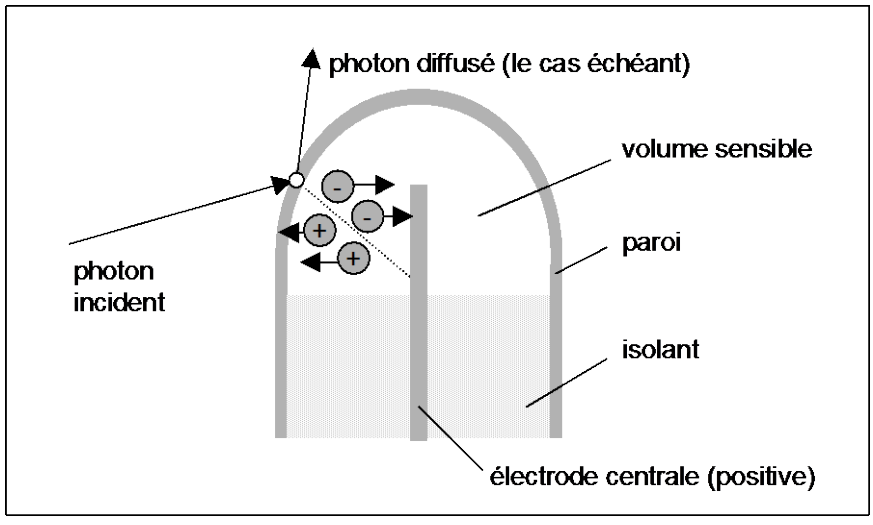


Figure 2.6: Typical diagram of an ionization chamber used in dosimetry for radiotherapy beams.

2.3. Proportional counter

When the voltage applied between the electrodes increases again, the initially created charges are sufficiently accelerated to subsequently induce ionizations when the electric field increases near the anode (Figure 2.7). This multiplication phenomenon is used in the *proportional counter* whose gain is a function of applied voltage.

The secondary electrons thus created can generate ultraviolet radiation in the gas or in the anode of the detector. This radiation can then be absorbed by the detection gas and generate parasite impulsions (see Figure 2.8). Parasite impulsions are minimized by not applying a too-high voltage, by choosing a gas which will absorb ultraviolet photons without excitation (methane, for example) or by choosing an anode material with a high activation energy.

A proportional counter generally functions in impulse mode. In this way it does not measure the charge traveling through the system but counts the number of electrical impulses produced by the detected radiation. Being able to count impulses individually explains the high sensitivity of this type of detector. This detector also makes it possible to discriminate between particles by the quantity of energy deposited in the counting gas. The distinction between α and β particles might occur because the amplitude of the impulse produced by an α particle is greater than the amplitude produced by a β particle.

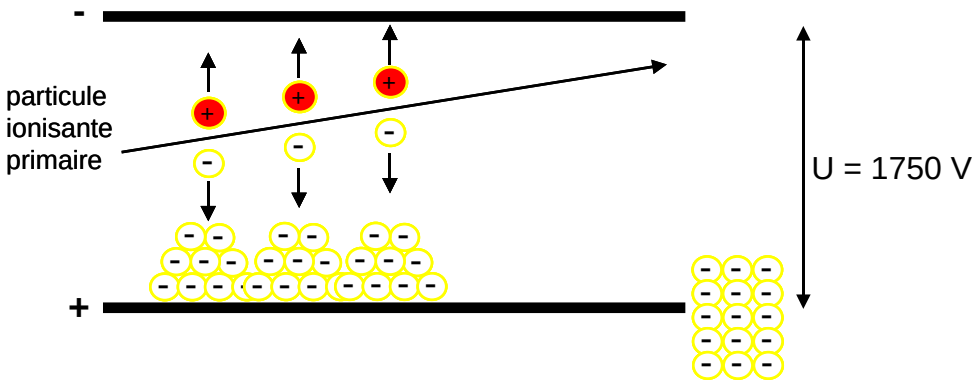


Figure 2.7: Functioning in proportional counter mode. Each electron created by the primary ionizing particle creates on its own an ensemble of secondary charges near the anode.

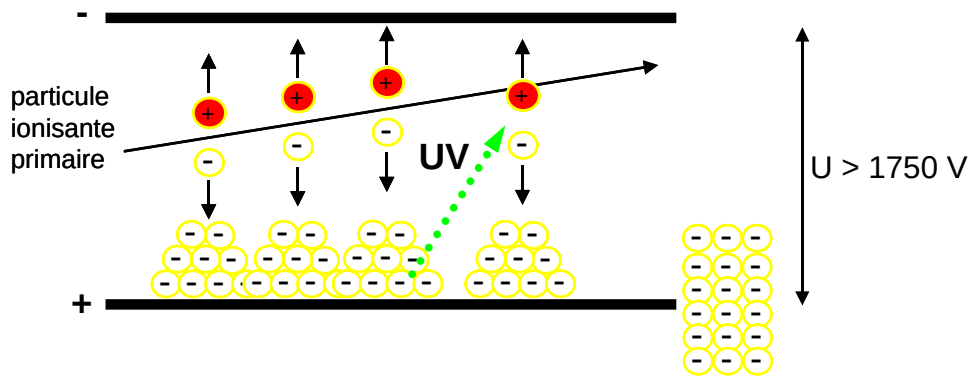


Figure 2.8: Production of parasite impulses in a proportional counter following the production of ultraviolet radiation created through the excitation of secondary electrons.

This detector is used for measuring α, β and γ radiation contamination. For this, the sensitive volume which can be filled with xenon or a hydrocarbon mixture generally presents a large surface of detection delimited by an adequately thin window (see example in Figure 2.9). This detector is also used for measuring external radiation in the field of radiation protection, in particular for high dose rate areas where the Geiger-Müller counter (see below) becomes saturated.

(a) 

(b) 

Figure 2.9: (a) Example large surface proportional counter used to measure surface contaminations (Berthold LB 1210). (b) Example proportional counter used to monitor foot and hand contamination (Berthold LB 1041 counter).

2.4. Geiger-Müller counter

When the electrode's voltage is increased even further, we enter zone E of Figure 2.3, where each event sets off the same avalanche of charges, no matter the energy of the incident particle (see Figure 2.10).

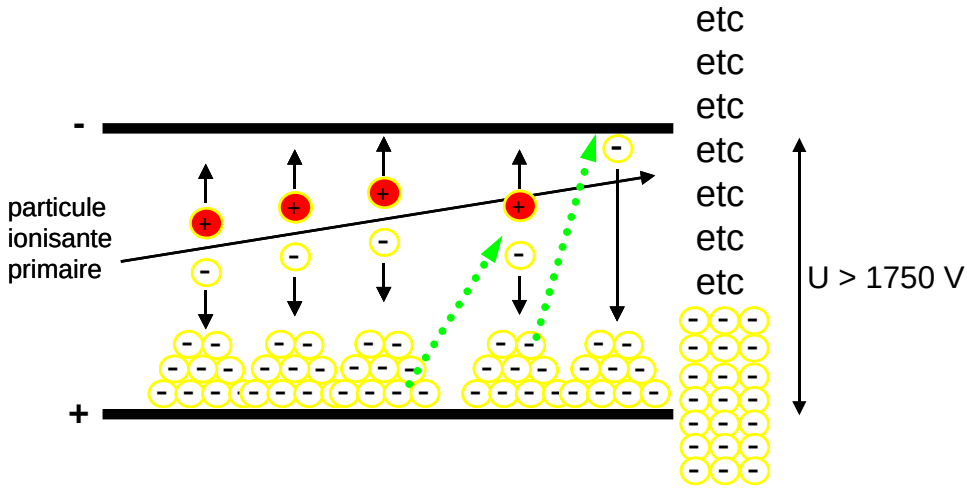


Figure 2.10: Functioning in Geiger-Müller mode. Each primary ionizing particle produces the same avalanche of secondary electrons through generated UV rays.

Instruments which work in this high-voltage zone are called *Geiger-Müller* counters (GM). To prevent the avalanche phenomenon from transforming into a continuous discharge, the filling gas is maintained in a sealed volume with a quenching agent. Rare gases like argon or helium with quenching agents comprised of methane or ethanol are frequently used. The impulse rate

measured by a GM can only be linked to a dosimetric parameter with well-defined measuring conditions (geometry, radiation quality, etc.).

A GM counter is generally used as a γ radiation detector in radiation protection (see the example in Figure 2.11). However, when equipped with a thin window (Mylar), it can be used to detect α or β radiation.

Figure 2.11: Example of a Geiger-Müller counter used in radiation protection (Berthold LB 1236 detector).

2.5. Semi-conductor detector

At the junction of two semi-conductor crystals p and n, there is a space without charge carriers (electrons, holes) which is submitted to an electric field. The irradiation of this zone produces, like in an ionization chamber, electrical charges whose collection allows for measuring the dose or activity of a radiation source.

This system is used when manufacturing small-volume (some mm^3) measuring probes for dosimetry, in particular for the field of radiodiagnostics (see Figure 2.12). On the other hand, germanium detectors, which have a relatively large sensitive volume (~ 100 cm^3) have excellent energy resolution and are used for precision spectrometry. These detectors work at the temperature of liquid nitrogen.

Figure 2.12: Example of radiodiagnostic dosimetry including a semi-conductor detector (Wellhöfer Dosimax).

3. Luminescence-based detectors

Certain materials, called luminescents, emit light when they are irradiated. Since the quantity of light emitted is directly linked to certain irradiation parameters, these materials are used as detectors to measure the dose or activity of radiation sources.

3.1. Luminescence: fluorescence and phosphorescence

Luminescence is the emission of light from a material having absorbed energy from an external source like ultraviolet or high-energy radiation. This light emission is more easily understood through a description of the energy state of the surrounding electrons. An electron is considered to be in its *fundamental* state when it is at its lowest energy level; this is a stable state. Absorbing external energy can put the electron into an unstable *excited* state in which it will only remain for a short moment before falling back into its fundamental state by emitting a photon with the same amount of energy as the difference between the excited state and the fundamental state. This phenomenon is called *fluorescence*. *Phosphorescence* is different from fluorescence in the fact that instead of returning directly to its fundamental state, the electron can also fall into a *metastable*[3] state of intermediate energy and stay there for a certain time (see Figure 2.13). The molecular agitation of the material, or an external energy contribution in the form of heat or laser radiation for example, can make the electron move from a metastable state to an excited state. The electron can then fall again into a metastable state, or fall into the fundamental state by emitting a photon having the same energy as the different between the excited state and the fundamental state.

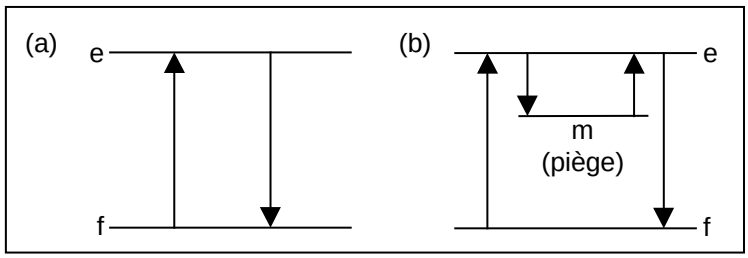


Figure 2.13: (a)Excited state e and fundamental state f showing absorption and emission in a fluorescence process. (b) Metastable state m (also called trap) occasioning a delay between excitation and emission in the phosphorescence process.

We know that a fluorescent emission is characterized by a short time period (length of time between the moment when the radiation is applied and the moment of luminescent emission) while phosphorescence is characterized by a long time period. To set the scale of the parameter, we can say that a time period lower than 10^{-8} s is certainly a fluorescence process while a time period longer than several seconds is most likely a phosphorescence process. Between these two extremes can be found both processes.

3.2. Luminescence and radiation measurements

Materials with luminescent properties can be used to conduct ionizing radiation measurements. In this case, the excitation of the surrounding electrons occurs via secondary electrons produced through the interaction of radiation in the material. Fluorescent materials are used for direct radiation measurements like liquid or solid scintillation. The main application of phosphorescence is for thermoluminescence dosimetry (TLD) in which the energy stored by the material after irradiation is released using heat.

3.3. Solid scintillators

A solid scintillator is generally used to detect photons through photon-matter interactions (photoelectric and Compton effect, pair creation). The electrons produced through these primary interactions are slowed down by ionizing and exciting the scintillator's material. The scintillator's excited electrons then return to their fundamental state by emitting light.

[3] The metastable state is also often called "trap".

A detector based on a solid scintillator is able to gather the energy of the visible light photons produced inside the scintillator and transform them into electrical current. This transformation occurs in two stages. First, the visible light photons, confined into a light-proof area, are converted via the photoelectric effect into electrons using a photocathode. At the exit of the photocathode, a photomultiplier, comprised of a series of electrodes (from 10 to 15) called dynodes, brought to increasing voltage levels, amplifies the electrons. The electronic emission of the photocathode is thus multiplied by a factor of about a million; this is called the photomultiplier gain (see Figure 2.14).

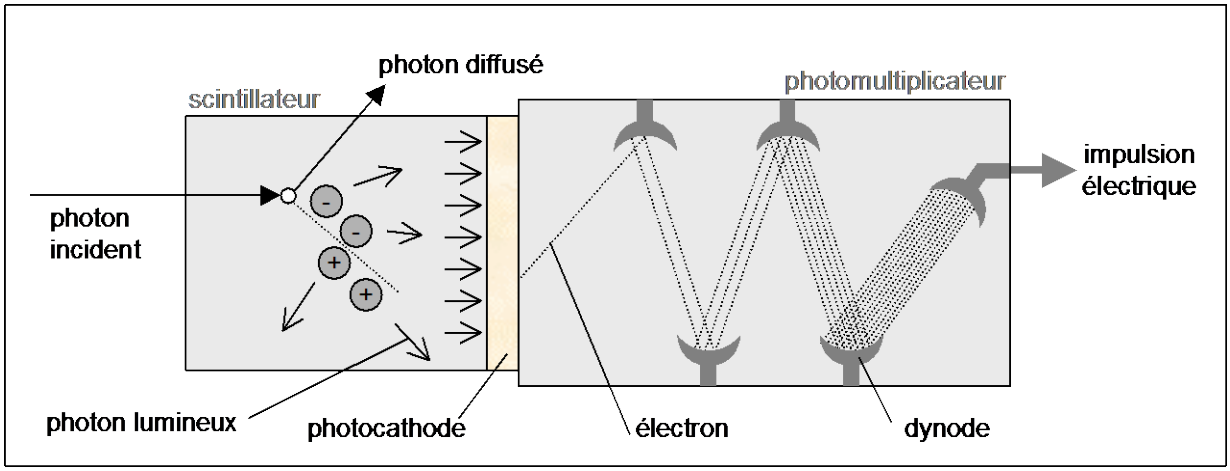


Figure 2.14: How a scintillation detector works.

The measurement electronic can function either in continuous mode (average current delivered to the exit of the photomultiplier) or in impulse mode. The response of a scintillation counter varies greatly with the radiation energy. It is, however, extremely sensitive.

When the electronic device functions in impulse mode, the scintillation can be used for a very important application: spectrometry. In this case, the impulses have an amplitude proportional to the energy absorbed in the crystal and by adding a multichannel analyzer you can “sort” and “class” impulses according to amplitude. The curve obtained is the energy spectrum of the radiation, characteristic of the radiation source being considered. This technique is used not only to identify, but also to discriminate between several elements present, or, if necessary, determine the activity of a radionuclide.

However, the spectrometric resolution of scintillators is much poorer than semi-conductor detectors. Figure 2.15 shows that the peaks obtained with the semi-conductor detector are much clearer than those obtained with the scintillating detector (NaI).

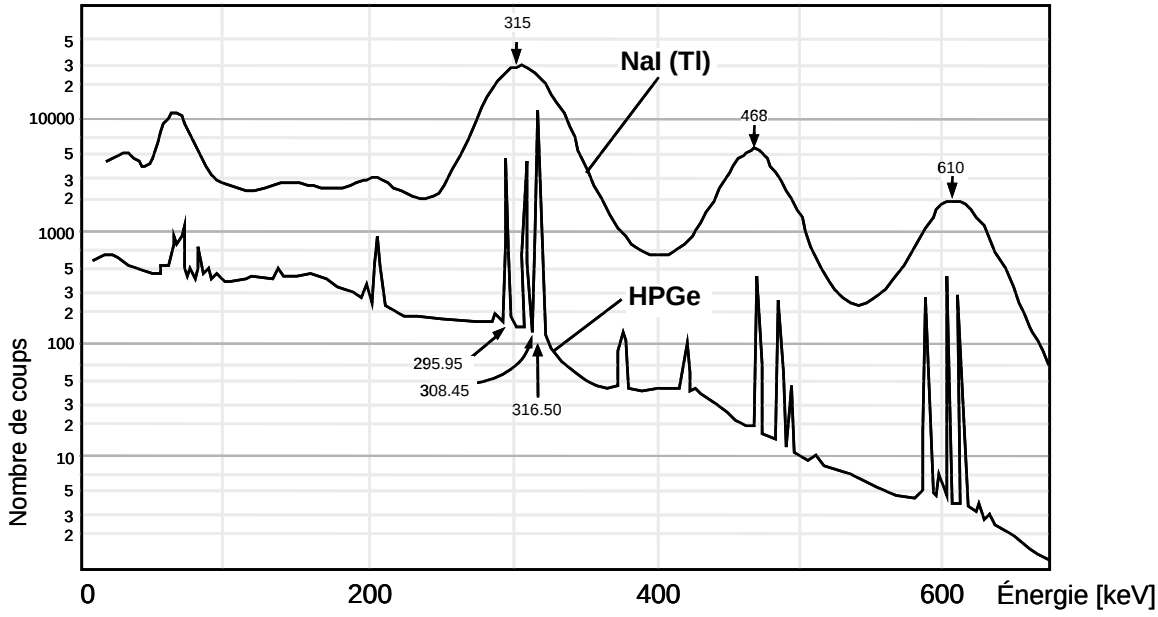


Figure 2.15: Comparison of Iridium-192 spectrum measured with a crystal scintillator (NaI) vs. a semi-conductor detector (HPGe).

3.4. Liquid scintillators

Liquid scintillation counting is used in the laboratory to measure beta emitters (as well as alpha). The sample is dissolved or placed into a suspension in a "cocktail" containing an aromatic solvent and a small amount of fluorescent additive. The beta particles emitted by the sample transfer their energy to the solvent which in turn transfers its energy to the fluoride. The excited fluoride molecules then rapidly dissipate their energy by emitting light.

Samples are generally placed in a small transparent or translucent vial (often glass or plastic) which is loaded into an instrument called a liquid scintillation counter. The counter generally contains two photomultiplier tubes recording in coincidence in order to only record the signals appearing simultaneously on both tubes. This helps eliminate background signals which only affect one tube at a time.

In ideal conditions, counting efficiency typically varies between 30% for Tritium (low beta energy) and 100% for P-32 (high beta energy). Certain chemical compositions (like chloride compositions) or significant coloration of the sample can interfere with the counting process. These interferences, known as quenching can be taken into consideration by conditioning the sample or treating the data.

High-energy beta emitters, like P-32 can also be measured in a scintillation counter without using a cocktail. This technique, known as Cherenkov counting, is based on Cherenkov photonic radiation emitted when a particle moves faster than the speed of light in matter. This type of counting is generally reserved for initial quick measurements which are not extremely precise.

3.5. Thermoluminescent dosimeters (TLD)

A thermoluminescent dosimeter (TLD) is an isolating material which, when exposed to radiation, stores a part of the dissipated energy by maintaining electrons in a metastable state for a long period of time. The electrons may be forced back to their fundamental energy state by submitting the TLD to a heat source. This return provokes a liberation of energy in the form of light photons (phosphorescence) whose number can be directly linked to the dose delivered by the radiation. Figure 2.16 describes exactly how this dosimetry works.

The simplicity of the equipment needed to conduct thermoluminescent measurements is such that a precise, reliable system can be established at a minimal cost. Necessary instrumentation: an oven, a good temperature control mechanism, a photodetector and the thermoluminescent element itself.

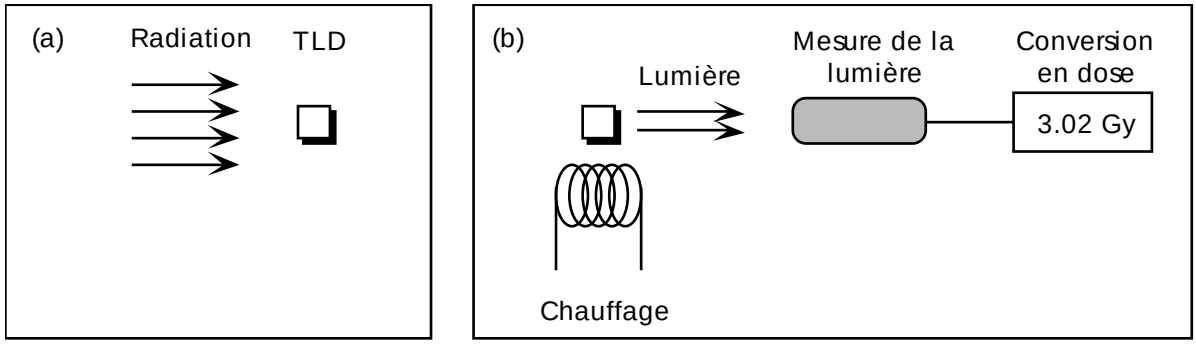


Figure 2.16: Diagram explaining how TLD works. (a)The TLD is irradiated. (b) The TLD is read by measuring the quantity of light emitted when heated. The quantity of light emitted can be directly linked to the dose received by the TLD.

4. Calorimetry-based detectors

Most of the energy delivered to matter through radiation is transformed into heat. A direct measure of this heat is another way to make an absolute determination of absorbed dose. However, this method is particularly difficult; in essence, the temperature increase corresponding to an absorbed dose of 1 Gy is approximately 10^{-3} °C in graphite. This measurement technique, which allows for an absolute determination of absorbed dose, is mainly used in national metrology labs to calibrate other measuring instruments (see Figure 2.17).

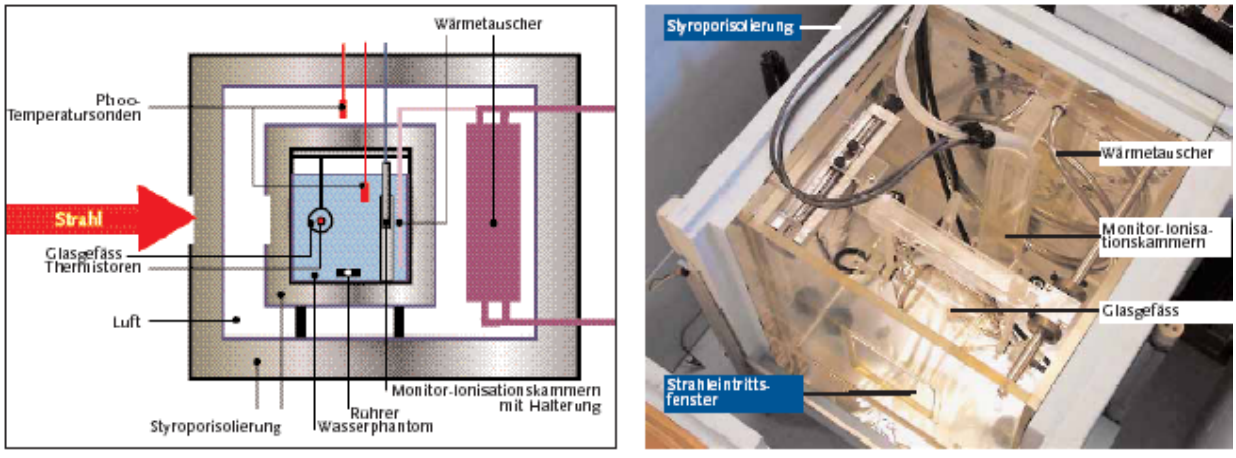


Figure 2.17: Diagram of calorimetry in water performed at the Swiss Primary Lab: METAS.

5. Chemical detectors

5.1. Photographic emulsion

The effect used in this type of detector is the blackening of film expressed through optical density. The film's response to dose is non-linear and varies greatly with γ radiation energy.

Photographic emulsion is still used in conventional radiodiagnostics. This technique is also interesting because it highlights the microdivision of dose, for example for the autoradiography

of material containing radioactive substances, or for qualifying the dose distribution in a radiotherapy beam. Figure 6.18 presents the first radiography conducted by Röntgen.

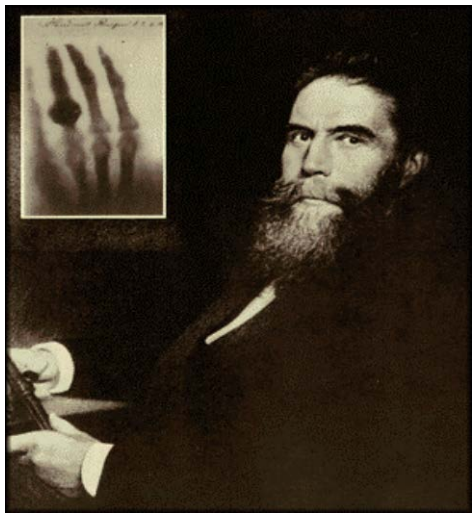


Figure 2.18: First radiography taken by W.C. Röntgen in 1895 of his spouse's hand.

5.2. Fricke Dosimetry

Irradiating a ferrous solution (Fe^{2+}) leads to the appearance of ferric ions (Fe^{3+}) whose relationship is directly linked to absorbed dose. This property is used as the basis of a system called Fricke dosimetry and is used as a primary measurement of absorbed dose.

5.3. Gel-based dosimetry

The first gel-based dosimeters were created in the 1980s by inserting ferric ions into a gel. The transition from a ferrous to a ferric state modifies the magnetic moment of the metallic ion, thus reducing the relaxation time of the water-based hydrogen atom. This allows for a three-dimensional dose measurement using nuclear magnetic resonance imaging (MRI). Unfortunately, the ferrous and ferric ions very quickly diffuse after irradiation and so the routine use of this type of dosimeter was never made possible.

But the technique helped to develop gels containing a mix of monomers and acrylamid whose irradiation leads to the polymerization of the monomers by also modifying the magnetic moment of the molecule. These gels are much more stable than the ones used before and can be used to measure doses up to several grays with a precision around 4 - 8%.

Since the end of the 1990s, this type of gel has been used routinely for making 3D measurements in the field of radiotherapy (see Figure 2.19). Reading this dosimeter is usually done via MRI, but the modification of the structure of the gel is such that it is even possible to carry out a reading through visible light, ultrasound and even transverse tomography (CT).

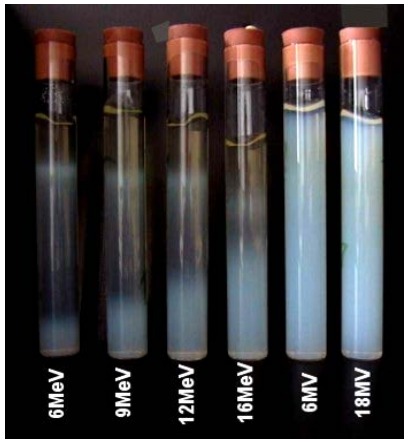


Figure 2.19: Example of polymer gels irradiated by photon beams at increasing energy.

5.4. Electronic paramagnetic resonance

Electronic paramagnetic resonance (EPR) is a spectroscopic technique which detects molecules having non-paired electrons (like free radicals, for example). The physical principle behind this method is the same as nuclear magnetic resonance. Instead of the nucleus spin, the electron spin is excited and then the relaxation signal is recorded.

The spin of certain materials, like ceramics, teeth and bones, is paired and does not produce an EPR signal. However, irradiation increases the proportion of non-paired spin and so the signal received then makes it possible to measure the dose received by the material in question.

This type of instrument makes an a posteriori measurement when no other kind of measuring instrument has been selected. Tiles were measured using this technique after the Chernobyl accident. Following the accidental irradiation of an individual, it is also possible to measure the signal produced by the person's bones or teeth.

A more regular application of the method is alanin dosimetry which functions along the same principle and appears in the form of a powder. The advantage of this type of dosimetry is its great linearity at very high doses.

5.5. Trace dosimetry

Allyl diglycol carbonate (known as CR-39) is a polymer commonly used in making eye glasses. It is used to measure neutron beams, since during a neutron-proton reaction, the recoil proton is slowed and creates chemical modifications along its trace. After exposure, these traces can be made visible by applying acid. The dose is then read either by visual counting or by using an automatic reader.

6. Bubble dosimetry

Bubble dosimetry is a technique which uses a tube filled with a transparent gel. This gel contains very fine drops of a superheated liquid. When any neutronic radiation touches one of the drops, the liquid turns into a vapor and forms a bubble which remains trapped inside the gel. This type of dosimetry measures neutrons in real time and is not sensitive to photons.

7. Activation measurement

Certain types of radiation, like neutrons, are relatively difficult to detect. An indirect way to measure them involves using a transfer material able to absorb neutrons and then emit a particle which is easier to detect. Example materials which can be used for this type of measurement are listed in Table 2.1

Table 2.1: Example material used to measure neutrons via activation. T corresponds to the half-life of the daughter atom produced. The threshold is the minimum amount of energy needed for neutrons to create a reaction.

Material	Reaction	T	Threshold [MeV]
F	$^{19}F(n,2n)^{18}F$	110 min	11.6
Mg	$^{24}Mg(n,p)^{24}Mg$	15 h	6.0
Al	$^{27}Al(n,a)^{24}Na$	15 h	4.9
Fe	$^{56}Fe(n,p)^{56}Mn$	2.6 h	4.9
Co	$^{59}Co(n,a)^{56}Mn$	2.6 h	5.2

8. Measuring through biological effect

As we will see further on in the text, ionizing radiation can modify the chromosomal structure of an irradiated individual. In a situation of strong irradiation, it is possible to estimate the received dose by calculating the number of chromosomal abnormalities in the entire cell structure. Figure 2.20 presents an example.

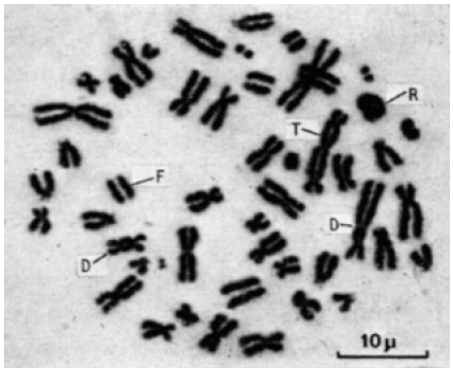


Figure 2.20: Example of chromosomes showing abnormalities. F : fragments; D : dicentric; T : tricentric; R : ring

9. Principle measurements required in radioprotection

The goal of all radiation measurements carried out within the framework of radiation protection is to evaluate the risks for exposed individuals in order to take the necessary steps to limit exposure. However, different situations may be encountered in which the appropriate parameters for describing “the amount of radiation” to which an individual is exposed are not identical. This is why it is extremely important to truly understand what type of measurement is most appropriate for the situations explained in the following paragraphs.

9.1. Monitoring of personal external exposure

The irradiation of personnel having occupational exposure to ionizing radiation is measured using a personal dosimeter, from month to month, or at a shorter interval when necessary. This monitoring is most often focused on whole body irradiation to X-rays or γ radiation, but also in certain instances to irradiation to the hands of individuals who generate preparations using open radioactive substances.

In the event that an individual not subject to occupational monitoring must enter a controlled zone, there is a specific measurement which can be conducted using a measuring instrument able to immediately display dose.

For locations with a much higher irradiation risk, a personal electronic dosimeter with an alarm may be needed.

Measuring instrument

Personal dosimeter, ring-dosimeter for hands, personal electronic dosimeter.

Parameters measured

Personal deep dose equivalent Hp(10) and personal surface dose equivalent Hp(0.07) in μSv or mSv.

Examples of practical situations

Monitoring of doses received by the personnel of a nuclear medicine department is guaranteed through personal dosimeters. Most often, thermoluminescent dosimeters are used in this situation.

Personnel working in a radiation room with high-rate beams wear a dosimeter with an alarm so they might be warned if the source becomes blocked.

9.2. Measuring ambient radiation

The goal of measuring ambient radiation is to determine individual irradiation in a relatively wide radiation field. This concerns penetrating radiation, meaning γ (X-ray or neutrons). It can also be used to determine the exposure linked to a specific task, or determine the efficiency of shielding protection (measure *without* shielding, then *with*). There are also stationary instruments which can be used for continuous monitoring of the level of ambient radiation in certain laboratories; these generally carry alarms and a data recording function.

Measuring instrument

Dose rate meter (the term *dosimeter* is most often used).

Parameter measured

Ambient dose rate equivalent in μSv/h or mSv/h and often also ambient dose equivalent in μSv or mSv.

Examples of practical situations

The irradiation of nuclear medicine personnel when manipulating Tc-99m for a radiodiagnostic exam can be evaluated using a dosimeter.

A radioactive package sent through the post must not present a dose rate higher than 5 μSv/h on the surface of the package. To ensure the shielding of a radioactive source is sufficient and that the limit is not exceeded, we can use a dosimeter when preparing the package for shipping.

9.3. Measuring surface contamination

Handling a radioactive substance in liquid or powder form can result in the diffusion of the radioactive material over a working surface, on instruments and the handler's hands, all of which introduces a risk of activity intake. This type of contamination can be managed by measuring surface contamination.

A direct measurement can be taken, or, if the direct radiation from the source is too high in the location to be tested, through a smear test.

Measuring instrument

Surface contamination monitor

Parameter measured

Activity of the radionuclide per surface unit in Bq/cm^2.

The only difficulty resides in the fact that the contamination monitors display the measurement result in "counts per second" or "impulses per second" (shown as "cps", "ips" or "s^{-1}"). Calculating the activity per surface unit from the counting rate is then done according to the sensitivity of the device for the radionuclide in question. This calculation is described in detail further on.

In practice, laboratories must have surface contamination monitors available which have been verified by a Swiss recognized verification Laboratory (IRA and PSI). The verification certificate indicates the counting rate for each radionuclide handled which corresponds to a surface contamination limit established by the Ordinance on Radiation Protection (ORaP). This facilitates user interpretation of the measurement.

Examples of practical situations

While administering a Co-57 product to a patient (orally) for a nuclear medicine exam, a part of the product is spilled on the floor. The amount and exact location of the contamination are determined using a surface contamination monitor. The instrument will then make it possible to verify the efficiency of decontamination procedures through measuring the cleaning towel and any contamination remaining on the floor.

Upon receipt of a radioactive package duly marked and containing a vial of I-125, a smear test of the vial is conducted to control whether any contents spilled during transport. The smear test is measured with a surface contamination monitor, away from the vial to avoid any radiation coming directly from the source.

9.4. Measuring activity intake (screening measurement)

Handling radioactive substances in liquid or powder form can involve a risk of activity intake (through inhalation or ingestion). For individuals exposed to this risk, regular monitoring is recommended. Monitoring consists in a rough measurement, called a screening measurement, whose goal is to detect any internal contamination. If necessary, a more precise measurement is then conducted with a dosimetry department for intake.

Measuring instrument

Depending on the incorporated radionuclide, different instrument types may be used (surface contamination monitor, dose rate monitor, etc.)

Parameter measured

Depending on the radionuclide incorporated, different parameters will be determined, generally in an indirect way (activity, specific activity, dose rate, etc.)

Examples of practical situations

The day after preparing a capsule of I-131, the lab worker checks the activity of his/her thyroid by placing a surface contamination monitor in front of him/her. If the net indication exceeds acceptable limits, the test is considered positive.

Testing intake of Tc-99m is conducted with a dose rate monitor placed on the abdomen. If the dose rate exceeds accepted limits, the control is considered positive.

9.5. Measuring air contamination

Measuring ambient radioactive contamination is necessary in working areas where radioactive gases or powdered products are used. On a professional level, this mainly concerns tritium in water vapor form. For the general population, radon and its decay products represent a non-negligible radiation exposure which can be determined by measuring their amounts in homes and living spaces.

Measuring instruments

Ambient Tritium counter, aerosol sampler, radon detector

Parameter measured

Activity per unit of air volume (Bq/m^3)

Examples of practical situations

In the radioluminescent painting industry, a continuous monitoring device for detecting ambient Tritium is installed in the workshops.

Radon detectors are available from various measuring authorities for private individuals wishing to test their living space. These detectors are small containers in which radon gas penetrates, creating a measurable effect in the sensitive material of the detector (ionizations, α particle traces, etc.)

9.6. Measuring activity of solid or liquid samples

In radiation protection, analyzing sample activity is essentially done to make statements regarding the disposal of solid or liquid waste. It is also used to test the internal contamination of individuals handling β emitter radionuclides (H-3, C-14, S-35, P-32, etc.). Urine tests are most common for this testing.

Measuring instruments

Scintillation detector (NaI) and semiconductor detector (high-purity germanium, HPGe) for γ emitters.

Liquid scintillation counter for β+ emitters.

Parameter measured

Activity (Bq) or specific activity of the sample (Bq/kg, Bq/l)

Example practical situation

The retaining vat for waste water from a laboratory working with open radioactive sources must be cleaned. Before the procedure, 1 liter of water is removed and analyzed with an HPGe semiconductor detector to determine whether the activity of the water will allow it to enter the sewer system.

10.Important properties of a measuring instrument

10.1. Sensitivity

In practice, we are used to see a measuring instrument directly display the value of the parameter measured. In the field of radiation measurements, however, this is not always the case and the instrument's indication may be displayed in raw form. The parameter measured is obtained in this case via the sensitivity *S* of the measuring instrument, by taking the ratio of the displayed value (often the counting rate) and the sensitivity. Two examples below will make this more clear.

Activity measurement

$$A\left[Bq\right] = \frac{\dot{N}_{net}\left[cps\right]}{S\left[cps/Bq\right]} \quad , \qquad (0.16)$$

where *A* is the activity and $\dot{N}_{net}$ is the net counting rate (after subtracting background noise)

Measuring surface contamination

$$A_s\left[Bq/cm^2\right] = \frac{\dot{N}_{net}\left[cps\right]}{S\left[\frac{cps}{Bq/cm^2}\right]} \quad , \qquad (0.17)$$

where A_s = surface contamination and $\dot{N}_{net}$ = net counting rate (after subtracting background noise).

Sensitivity expresses the ability of the instrument to detect radiation. The greater the sensitivity value, the better the instrument will work for the measurement in question. Note that sensitivity increases with the size of the detector.

We also very often use the calibration factor $F_{étal}$ of the measuring instrument. This is the inverse of sensitivity ; $F_{étal} = \frac{1}{S}$.

10.2. Background noise

Background noise is the measurement instrument's residual signal in the absence of radiation or in the presence of only natural radiation.

Example. The typical background noise of a surface contamination monitor based on a proportional counter of 10 cm x 10 cm is on the order of 10 cps (caused by natural radiation). You must subtract background noise from the displayed value of the counting rate to obtain the net counting rate caused by contamination:

$$\dot{N}_{net} = \dot{N}_{brut} - \dot{N}_{bdf} \quad , \qquad (0.18)$$

10.3. Detection limits

Detection limit is the smallest quantity of radiation which can be detected by the measuring instrument. This value is linked to the instrument's sensitivity and background noise.

10.4. Response as a function of energy

The response as a function of energy is the variation of the value indicated by the measuring instrument as a function of the energy of the radiation, for a set quantity of radiation. Figure 2.21 presents the energy response curves of two dose rate monitors, one Geiger-Mueller counter and one ionization chamber.

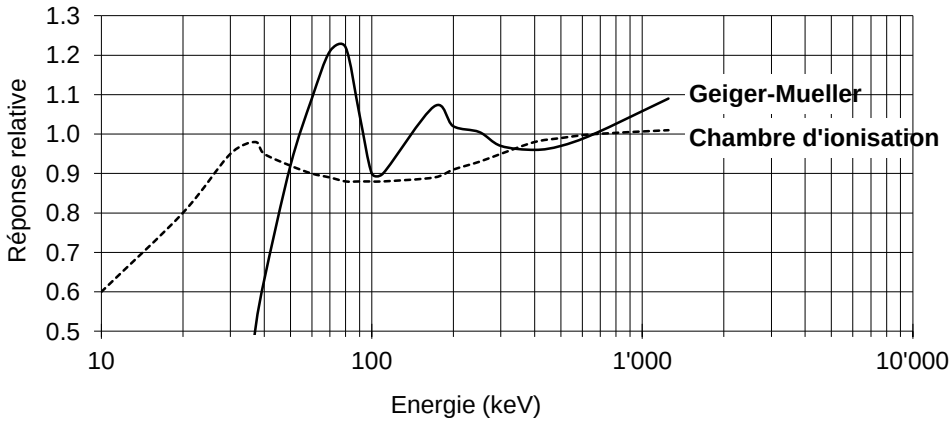


Figure 2.21: Energy response curves of two dose rate monitors.

The energy response of a Geiger-Mueller counter is generally less uniform than an ionization chamber, in particular at low-energy where we can see a drop in response below 50-60 keV.

10.5. Directional response

The directional response is the variation of the value indicated by the measuring instrument as a function of its orientation with respect to the direction of the incident radiation, for a set quantity of radiation.

10.6. Thickness of the detector's entry window

The sensitive volume of the counter (counting gas, scintillator) is always closed within a confined area (gas detector), protected from light (scintillation detector) or mechanically protected. This area is made up of a certain thickness of material that the radiation must pass through to be detected. Non-penetrating radiation (low-energy β and mostly α) are only measurable with instruments having a thin window (see Table 2.2).

Table 2.1 : Example of typical window thicknesses for low-energy radiation sources.

Radiation source:	Window thickness:
C-14 (β at 156 keV)	~ 5 mg/cm^2
α emitters:	< 1 mg/cm^2
H-3 (β at 18.6 keV)	will not cross through any window thickness!

11.Legal requirements for radiation protection instrumentation

According to the Ordinance on Radiation Protection (ORaP), laboratories are subject to legal requirements concerning their radiation protection measuring instruments. This is meant to guarantee the availability and measuring precision of said instruments.

Dose rate monitors must undergo legal verification every 3 years with a service recognized by the Swiss Federal Department of Metrology METAS (IRA or PSI). Verification involves checking that the instrument is suitable for the use it is designated for and verifying whether the indicated value is correct, at the nearest tolerance.

In the same way, surface contamination monitors must be verified within the same time period and by the same services. Verification involves establishing, for the radionuclides handled by the requesting laboratory, the counting rate indicated by the instrument in the presence of a contamination equal to one times the CS guideline value (legal limit).

Instruments used in screening measurements for intake must also be verified or calibrated by a recognized service.

Summary:

- *Detectors based on ionization are the ionization chamber, the proportional counter and Geiger-Müller counter.*
- *The signal in an ionization chamber is caused by the primary ionization produced by the charged particles. This detector allows for the instant measurement of relatively high dose rates (applicable to radiotherapy).*
- *In the proportional counter, the signal is proportional to the primary ionization produced by the charged particles (amplification by production of secondary ionizations). This detector, in impulse mode, is extremely sensitive. Its main applications are for measuring alpha and beta/gamma contamination and measuring external radiation fields.*
- *In a Geiger-Muller counter, each radiation occasions an avalanche of charges. This is a sturdy counter used in radioprotection.*
- *In a semiconductor detector, radiation produces charges in a zone free of any charges. The main uses of semiconductor detectors are small volume probes and γ spectrometry.*
- *In a scintillation detector, the signal is linked to the quantity of light produced. This signal is generally amplified in a photomultiplier. The main applications of scintillation are measuring contamination, measuring low activity sources, measuring α and β emitters in liquids (liquid scintillation) and spectrometry.*
- *In thermoluminescent dosimetry, some of the electrons produced during ionization are trapped beneath crystal impurities. When the dosimeter is heated, electrons are freed and emit light when they recombine with the atom of the crystal. The main applications of thermoluminescence are measuring patient doses in radiotherapy, personal dosimetry for workers with occupational exposure to radiation and measuring environmental dose.*
- *Fricke chemical dosimetry is based on the transformation of ferrous iron (Fe^{2+}) into ferric iron (Fe^{3+}) in an aqueous solution.*
- *The sensitivity of photographic emulsion is characterized by a non-linear response. Its main application, beyond radiodiagnostic, is autoradiography.*
- *In calorimetry, we measure the energy deposited by radiation and transformed into heat. The method is not very sensitive, but absolute. It is used in national labs for calibrating other measurement instruments.*
- *Personal dosimetry measures the deep dose and surface dose of individuals with occupational exposure to ionizing radiation.*
- *A dose rate monitor is used to evaluate the external radiation field. It indicates the deep dose rate.*
- *A surface contamination monitor is used to determine contamination. It indicates a counting rate which is then interpreted using the data from the instrument's verification certificate.*

12. Practice questions

1. The mean energy expended in the air per ion pair formed is 34 eV. Calculate the charge produced by a 10 keV electron in an ionization chamber.

2. The mean energy needed in a semiconductor for one ionization is 3.5 eV. Calculate the charge produced by a 10 keV electron and compare the results with the answer from question 2.

3. If we accept that the resolution of a scintillation spectrometer is 8% (FWHM: Width to mid-height), can we separate the two lines of Cobalt-60 (1170 keV and 1330 keV) with this system?

CHAPTER • 3

OPERATIONAL RADIATION PROTECTION

Course goals
• *Cite secondary limits and guideline values*
• *Describe protection methods against external irradiation.*
• *Describe protection methods against internal irradiation.*
• *Understand the necessary basics for preparing the decontamination of material.*
• *Be able to manage radioactive sources and waste.*
• *Be informed about problems linked to the transport of radioactive substances.*
• *Use Index 3 of ORaP for special situations.*

1. Secondary limits and guideline values

1.1.1. Introduction

Primary limit values (those which engage a fixed limit in the ordinance) are dose equivalents to organs $H_{T,}$ particularly to lens of the eye, the skin and the extremities, as well as effective dose E. Since it is not possible to directly measure dose equivalents to organs, nor the effective dose, secondary limit values, called operational parameters, were defined. For ambient dosimetry, this is the ambient dose equivalent H*(10) and the directional dose equivalent H' (0.07). For personal dosimetry, the personal deep dose $H_p(10)$ and the personal surface dose $H_p(0.07)$ are used for external radiation, and the committed effective dose E_{50} is used for internal irradiation. All these parameters were described in detail in Chapter 5.

Furthermore, for radiation protection needs, two secondary activity limits and two guideline values were established:

- exemption limit, LE
- authorization limit, LA
- guideline value for air contamination, CA
- guideline value for surface contamination, CS.

Below, we present the role of these parameters and the way in which they are calculated from other parameters.

1.1.2. Exemption limit

The exemption limit applies to specific activities of solids and liquids. Below the exemption limit, the ordinance is no longer applicable. The following situations are an exception:

- water contamination; ingested quantities might be significant; for this situation, the ordinance sets the specific contamination limit at 1/100 of the LE;

- usual objects; here also ORaP uses 1/100 of the LE;
- food stuffs; particular limits are fixed by the Ordinance on foreign substances and ingredients of food products.

Models were developed which take into consideration the following paths of irradiation:

- whole body external irradiation; accepted that the entire environment is contaminated at LE/1000;
- skin irradiation upon contact with substances; calculated for dilution to a factor 1000 and a surface contamination corresponding to the activity contained in a density of 1 $g.cm^{-2}$;
- inhalation of a substance; calculated for a dilution of a factor 1000 and to a contamination through dust corresponding to the Maximum Acceptable Concentration (MAC) limit for workers;
- ingestion of the substance; calculated for a dilution of a factor 1000 and an ingestion of 1000 kg of the substance per year.

Of these four possible sources of irradiation, the most constraining is ingestion and so it is retained in the ordinance. The limit of effective annual dose was set at 10 µSv.

> And so LE, the exemption limit for a specific activity, in case of ingestion of 1 kg of the substance, leads to,
> a committed effective dose of 10 µSv.

LE can be deduced from e_{ing} using the following equation:

$$LE\left(Bq/kg\right) = \frac{10^{-5}\left(Sv \cdot kg^{-1}\right)}{e_{ing}\left(Sv \cdot Bq^{-1}\right)}.$$

1.1.3. Authorization limit

Authorization limit is the activity of a substance above which an authorization is needed to handle it as an open source. The major risk here is inhalation and the limit for a one-time inhalation of the total handled quantity must not exceed 5mSv.

> And so the LA, the authorization limit, for an inhaled substance, is the same as,
> a committed effective dose of 5 mSv…

LA can be deduced from e_{inh} using the following equation:

$$LA\left(Bq\right) = \frac{5 \cdot 10^{-3}\left(Sv\right)}{e_{inh}\left(Sv \cdot Bq^{-1}\right)}.$$

There are situations in which the value of LA calculated with the model above would be lower than LE. In this case, we take the LE value for LA.

1.1.4. Guideline value for air contamination

This parameter (AC) is calculated so that one person, working the entire year (40 hours per week and 50 weeks per year); breathing 1.2 m^3/h) in air contaminated at the guideline value, receives a committed effective dose equal to the annual occupational dose limit (20 mSv). AC can be deduced from e_{inh} using the following equation:

$$CA\left(Bq\cdot m^{-3}\right)=\frac{8,33\cdot 10^{-6}\left(Sv\cdot m^{-3}\right)}{e_{inh}\left(Sv\cdot Bq^{-1}\right)}$$.

1.1.5. Guideline value for surface contamination

This is the accepted limit value outside controlled zones. It is calculated so that:

- a repeated skin irradiation for the entire year does not exceed 50 mSv per year;
- that daily ingestion of the contamination from a surface area of 10 cm² does not lead to an effective dose higher than 0.5 mSv/year.
- that a one-time inhalation of 10% of the activity corresponding to 100 cm² does not lead to a dose higher than 0.5 mSv/year;
- a maximum value of 1000 Bq.cm^{-2}.

The SC value is established according to the following equation:

$$CS\left(Bq\cdot cm^{-2}\right)=\min\begin{cases}\dfrac{5,71}{h_e\left(0,07\right)\left(mSv\cdot h^{-1}\cdot kBq^{-1}\cdot cm^{2}\right)}\\ \dfrac{2,5\cdot 10^{-7}}{e_{ing}\left(Sv\cdot Bq^{-1}\right)}\\ \dfrac{5\cdot 10^{-5}}{e_{inh}\left(Sv\cdot Bq^{-1}\right)}\\ 1000\ Bq\cdot cm^{-2}.\end{cases}$$

CS values are rounded to the half-decade.

2. Protection methods against external irradiation.

The overall dose received by an individual near a radioactive source can be calculated using the following equation:

$$"H"="A"\cdot\frac{"h"}{r^2}\cdot t\cdot T$$

with "H" : parameter characterizing the dose received by an individual (H_T, E, H_{10}, $H_{0,07}$, ...);
"A" : parameter characterizing the importance of the radiation source; this is the activity of the source, the quantity of radiation produced by an accelerator, etc;
"h" : "dose" constant indicating the "dose" per unit of intensity of the source;
r : the distance between the individual and the radiation source;
t : duration of irradiation;
T : transmission of radiation through a screen placed between the source and the individual.

From this equation, we take the four rules of protection against external radiation:

- source selection (focus on "h" and "A");
- increase the distance;

- reduce exposure time;
- proper positioning of a screen.

2.1. Source selection

Selecting a source is based on the intensity of the source (focus on “A”) and the characteristics of the emitted radiation (focus on “h”).

Protection by reducing the intensity of the source is obvious, which is why the activity of the radiation source used will always be reduced to the necessary minimum for any planned manipulation. For instance:

- immediate storage of the parent solution following a dilution for use in chemistry or biology;

For any application, use the least penetrating radiation possible which is compatible with the application goal. When choosing a radionuclide in particular, favor one with minimal values for “h" parameters. Also try to avoid any parasitic penetrating radiation, in other words radiation which isn’t useful for the application in question. Here are some examples:

- use pure β sources when only interested in β radiation (thickness measurement scheme, scintillation source, etc) ;
- use Iodine-125 (35 keV γ radiation) rather than Iodine-131 (360 keV γ radiation) for biological marking.

2.2. Reduce working time

This is a simple, efficient and economical measure which is not used as often as it should be. Prior practice of the operation without the source is a good way to reduce irradiation for the worker by increasing operation rapidity and eliminating any useless gestures, dead time and unexpected occurrences.

2.3. Maintain maximum distance

Dose rate evolves as the inverse of the square of the distance to the source, and so the user should always keep as far from the source as possible. At 1 m from a given source, the rate is already approximately 1/10,000 of its value at 1 cm. This is especially important at the first few centimeters; for this same reason, using tweezers is also extremely effective in terms of limiting dose received to the hands when manipulating radioactive sources.

Note that when the source is extended (surface contamination or activated objects), the dose diminishes less quickly than the square of the distance in areas near the source.

2.4. Use protective screens

Absorption of radiation by placing screens between the source and the user is a common practice in radiation protection. In this context, the radiation’s ability to penetrate the selected screen is what must be considered.

For α radiation, the screen must be thicker than the particle range. Since α radiation is not very penetrating, the range of the particles in tissue does not exceed several dozen µm, which is lower than the corneal layer of the epidermis (70 µm) made up of dead skin cells. α radiation can be stopped using a simple sheet of paper. In this situation, the risk of external radiation is practically zero and protection focuses on controlling any risk of contamination and internal irradiation.

For β radiation, the range of the particles varies from several mm to about 1 cm in water (see Table 3.1). So the main characteristic of the screen is its density. The risks of external irradiation when manipulating β emitters are relatively higher than when manipulating α emitters. However, they are still fairly limited and installing an effective radiation protection is nearly always possible.

In these conditions, protection against β radiation having an energy higher than 150 keV is mostly done using a 1 cm-thick plastic screen. For high energy β radiation, the production of radiation through slowing in high Z materials should be taken into consideration. When the source is intense and has high energy, β radiation becomes non-negligible, even for weaker Z shielding materials. It is necessary to add lead protection to stop any radiation from slowing around the plastic shielding used to stop β radiation.

According to G. Sitzlack (Strahlenschutzpraxis, Verlag Tribüne, Berlin, 1974), it is possible to calculate the stopping radiation produced using the following empirical formula:

$$\dot{H} = 0,26 \cdot \frac{Z \cdot A}{r^2} \cdot E^2$$

in which $\dot{H}$: dose rate in mSv/h;
Z : atomic number of the material absorbing the electrons;
A : activity in GBq;
r : distance to the source in cm;
E : E_{max} of β in MeV.

Attention, this formula is only valid if all the β radiation is absorbed!

For γ radiation, attenuation is exponential. The screen is characterized by its transmission factor (T), which is the ratio of dose rate with and without the screen. For a given transmission factor, the thickness of the screen will depend on the energy of the γ radiation and the material. In Figures 3.1 and 3.2, the transmission of γ radiation from different radioactive sources is given as a function of the thickness of the screen for both lead and concrete.

When dealing with low energy γ radiation, using a material with a high atomic number is recommended because of its strong absorption capacity linked to the photoelectric effect.

Therefore, selecting a shielding thickness presupposes an “acceptable” residual dose rate behind the screen.

Neutron attenuation occurs in two stages: the first involves thermalizing the neutrons, the second absorbing them. Attenuation depends on the neutrons’ energy, the material and geometry. For Am-Be or Ra-Be type sources, the layers of half-attenuation at the tenth level are on the order of 20 cm for hydrogenated materials (polyethylene, paraffin) and on the order of 30 cm for concrete.

Table 3.1: Characteristics of several commonly-used β emitters

Radionuclide	^{3}H	^{14}C	^{35}S	^{45}Ca	^{32}P	^{90}Sr
Half-life	12.3 y	5730 y	88 d	165 d	14.2 d	28.1 d
Maximum energy (MeV)	0.018	0.154	0.167	0.254	1.71	2.24
Average energy (MeV)	0.006	0.050	0.049	0.077	0.70	0.93
Range in air (cm)	0.60	30	30	60	600	870
Range in water (cm)	0.00052	0.029	0.032	0.060	0.80	1.1
Fraction transmitted through the layer of dead skin (0.007 cm)	---	0.11	0.16	0.37	0.95	0.97

2.5. Controlling ambient radiation

Taking regular measurements of ambient radiation is an indirect protective measure. This allows an individual to adapt his/her behavior to the level of surrounding radiation.

The rate of ambient radiation is controlled on three levels:

- using fixed monitors. These devices, generally equipped with an alarm, guarantee that the level of radiation remains acceptable. The data provided by these devices can be recorded if necessary;
- using portable measuring devices. These devices are permanently available in the laboratory and can be used when needed;
- using personal measuring instruments. These devices indicate the dose received by the individual and can be equipped with an alarm or a quick-read function.

The level of ambient radiation must be checked frequently in areas where large radiation sources are used or when the dose rate can change unexpectedly or uncontrollably.

Figure 3.1: Radiation transmission of concrete for various radionuclides ($\rho = 2.35$ g/cm^3)

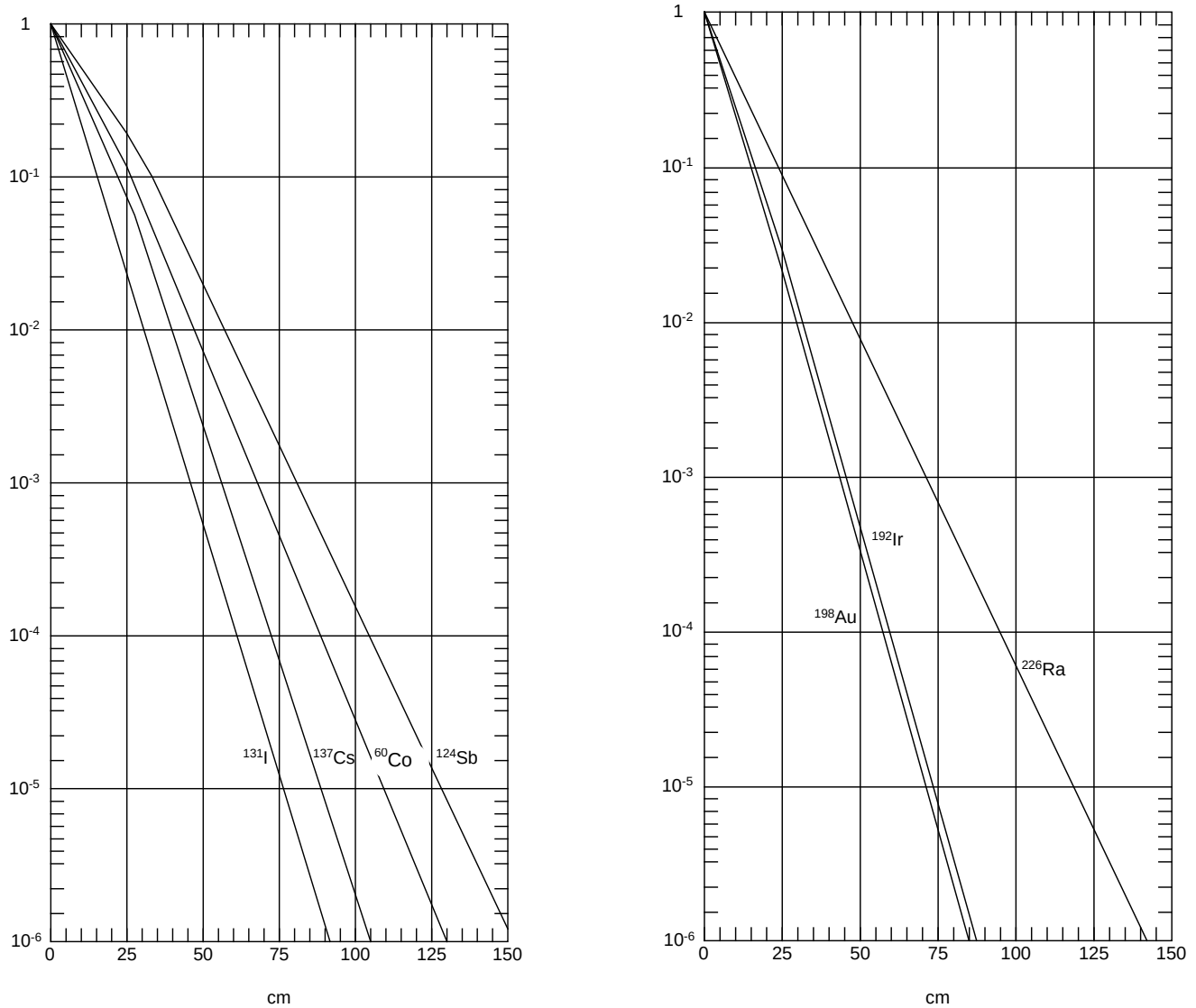
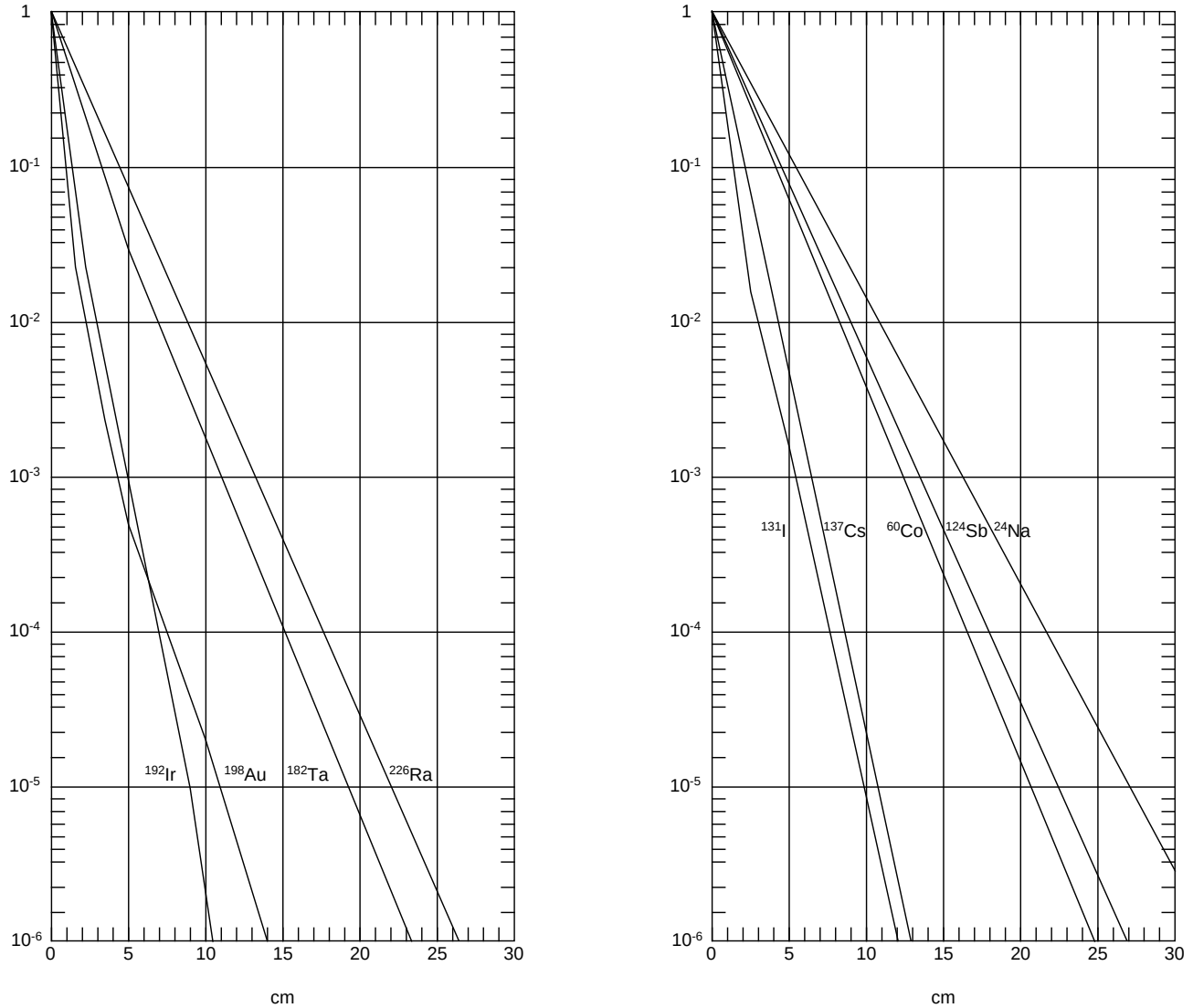


Figure 3.2: Radiation transmission of lead for various radionuclides ($\rho = 11.35$ g/cm^3)



2.6. Warning signs linked to ionizing radiation and marking of controlled areas

Controlled areas must be indicated by a warning sign. This sign must come with supplementary indications depending on the nature of the radiation sources.

Open radioactive sources

a. the most radiotoxic nuclide and its maximal activity;
b. classification of the working sector (type A, B or C);
c. the maximum degree of non-fixed surface contamination, in Bq/cm² or as guideline values for the nuclide in question;
d. the ambient dose rate in mSv per hour in the working area, if this is considered judicious;
e. information about necessary protective clothing as well as protection measures to take;
f. the warning sign.

Sealed radioactive sources

a. the most radiotoxic nuclide and the nuclide having the highest level of gamma radiation and its maximum activity;
b. the ambient dose rate in mSv per hour in the working area, if this is considered judicious;
c. the warning sign.

Installations generating ionizing radiation

a. the type of installation;
b. the type of the radiation (for example, electrons, X-ray, neutrons);
c. the ambient dose rate in mSv per hour in the working area, if this is considered judicious;
d. the warning sign.

3. Protection methods against contamination and internal irradiation.

Internal irradiation occurs following an internal contamination, meaning that the radiation source is present inside the organism.

3.1. Contamination risks

The risk of contamination linked to radioactive substances directly depends on their physical state: gas, aerosol, liquid or solid.

Gas and aerosols are considered a particularly high risk because of how easy it is to inhale them. Aerosols are rapidly trapped on any material found along their path: in particular, they attach themselves easily to dust particles. The latter then become strong contamination vectors. Which is why laboratories must be frequently cleaned.

Liquids pose a risk for surface contamination and atmospheric contamination if they are heated, agitated or spread.

Solids do not pose a significant contamination risk, except when they come in powder form or when they undergo transformation into an aerosol product (pulverization, fusion, put into solution).

As a general rule, protection against contamination consists in avoiding direct personal contact with radioactive materials and substances and limiting the pollution of material and lab

equipment to a strict minimum. However, it is important to remember that working with radioactive substances involves a certain unavoidable contamination.

The basic protection principles when working with open sources are the following:

- source selection;
- confining the source;
- protecting the individual to prevent contact and intake;
- adapting working methods.

3.2. Source selection

Like with external irradiation, selecting the source is based on both the characteristics of the source and its activity.

When working with an open source, we select the characteristics of the source according to the following criteria:

- half-life; we select the shortest half-life compatible with the planned application;
- type of emitter and radiation energy; we select the radioelement with the least dangerous radiation and the lowest energy;
- chemical forms; we prefer the chemical forms which are the most quickly eliminated by the body.

The dose factors for intake, e_{inh} and e_{ing}, which take all these parameters into consideration, facilitate an appreciation of the risk linked to using a given open source.

Reducing the activity of the source is also a very effective method of limiting radiological risk.

3.3. Protection through structures

Laboratories or working areas in which radioactive substances are manipulated are considered controlled areas in which working individuals may receive an effective dose higher than 1 mSv per year.

Ensuring protection through structures attempts to limit the risk or circumscribe the danger by confining the radioactive substance. This protection is guaranteed first by selecting the type of working area and its equipment. In Switzerland, this choice is stipulated by ORaP, in relation to the radionuclide and the activity used.

Annex 3 of ORaP indicates the authorization limit LA for each radionuclide. Table 3.2 gives the type of working area needed to match the activity manipulated.

Table 3.2: Types of working areas

Type	Maximum Activity
Normal	LA
Type C	100 x LA
Type B	10,000 x LA
Type A	Depending on authorization

The LA parameter is based on the risk of inhalation. When there is no risk, the supervising authority may set the type of working area on a case per case basis, depending on the existing risk of intake.

The requirements linked to the different types of working areas are established in the technical ordinance from the Federal Department of the Interior. They are summarized below.

Requirements linked to type C working areas are the following:

- laboratory must be labeled;
- F30 fireproof floors, ceilings and walls (windowed areas included);
- unbroken and impermeable floor coverings and working areas (plinths included).
- easily decontaminatable floors, walls and working areas;
- walls covered in washable paint;
- minimal furnishings;
- lockers for storing work clothes at the entrance;
- collecting containers for radioactive liquids or retaining vats;
- sink for hand decontamination;
- sink faucet and soap distributor which work without manual intervention;
- sufficient ventilation (artificial or natural);
- ventilation control system (hood, enclosed with window) to be used when producing vapor or dust;
- gas, water and electricity controls located outside the aspirator
- hand monitors located at the exit of the working area.

Type B working areas require the following additional requirements;

- T30 fire resistance for the doors;
- floor covering which climbs 10 cm up the walls;
- artificial ventilation with air filtration (5 changes per hour);
- control vats for dirty water (upon request of the supervising authority);
- cloakroom at the entry;
- foot and hand monitor at the exit.

Type A working areas require the following additional requirements;

- F90 fire resistance for the floors, ceilings and walls;
- mandatory waste water tanks;
- artificial ventilation with continuous depression and air filtration;
- cloakroom with showers or decontamination possibilities;
- working space monitoring and protection measures (may be automatic).

3.4. Means of personal protection

The goal of these particular working area structures is to confine a radioactive substance within a structure from which it cannot escape.

The goal of personal protection is the opposite, confine the user within a structure into which the substance cannot enter.

The minimum personal protection for manipulating open radioactive sources involves gloves and covering work clothes. For specific situations (accident, rare or high-risk manipulation), add respiratory protection (with special filters meant for the radionuclide used) as well as waterproof coveralls.

In certain situations, when the environment is particularly harsh, we can even confine the individual into a pressurized, sealed suit.

Also, hygienic masks (surgery masks, for example) or regular dust masks are to be avoided. They serve no purpose and provide a false sense of security.

3.5. Protection through working methods

This protection is guaranteed through a series of working rules, principles and techniques meant to decrease any risk of intake. In general, this means treating radionuclides as you would any toxic chemical.

The practical working methods in the laboratory should be studied on a case per case basis with respect to the principles of radiation protection discussed earlier. There are, however, a number of practical rules which should be applied overall:

- do not smoke, drink or eat in a laboratory where open sources are located;
- always use gloves when handling the source;
- take all necessary precautions to avoid punctures, cuts or other injuries;
- never use a pipette with the mouth;
- frequently monitor hand contamination;
- periodically monitor (and when finished with any work) surface contamination of working areas, shoes and clothing;
- never allow a contamination to sit about, wash hands as frequently as necessary;
- avoid working near the stock of a radioactive substance: put it away as soon as you've taken what you need for working;
- conduct any handling above a basin which can gather the substance if it disperses;
- prepare special containers for radioactive waste;
- label all radioactive products;
- monitor the contamination of any object leaving the working area;
- prepare response measures in the event of a contamination;
- in case of an accident, react calmly and carefully, avoid spreading the contamination, call upon the radiation protection officer.

Frequent monitoring of dose rates and contaminations in working areas is a way to check whether working methods are adequate. In certain situations, constant monitoring of air contamination in the working area may be necessary.

3.6. Decontamination

Any object which enters into contact with radioactive materials or with a contaminated surface may be contaminated. Decontamination is an operation meant to eliminate radioactive substances which have been deposited on working surfaces, objects or individuals. As a general rule, all contamination must be eliminated.

There are two types of contamination – fixed and non-fixed contamination. With fixed contamination, the radioactive substance is attached to the surface of the object and so presents no risk of dispersion. In non-fixed contamination, the contamination is transmissible to any object entering in contact with the surface. Decontamination measures focus on eliminating this second type of contamination.

3.6.1. Guideline values

The maximum allowable contamination is indicated in ORaP with respect to the contaminating radionuclide. These values, SC, are listed in Annex 3 and correspond to the maximum non-fixed activity allowable on surfaces outside controlled areas. They also apply to body contamination.

Inside controlled areas, surface contamination may be 10 x SC.

3.6.2. Measuring surface contamination

There are two methods of measuring contamination: direct and indirect. Direct measurements are taken with an appropriate instrument and measure the contamination of the object in question. We use the indirect method for measuring very low β radiation (^{3}H, for example) or in the presence of external radiation (measuring the external contamination of a recipient containing a source, in the presence of high background noise or fixed contamination). The indirect method consists of conducting a smear test using a filter paper or a damp cloth. The activity gathered on the smear test is measured and gives an indication of the state of the non-fixed contamination of the object in question.

3.6.3. Decontaminating objects and working surfaces

The contamination of working areas is directly linked to handling radioactive substances. These can be in the form of ions, non-charged combinations, or aggregates of submicroscopic, microscopic or macroscopic colloidal molecules. Fixation on surfaces depends on the size of the particles and the natures of the binding forces between the substances and the substrate. These forces can be electrostatic (charged particles), chemical (formation of chemical compounds) or mechanic (trapped in the grooves of the surface). In consideration of these different forces, decontamination procedures can be divided into three large categories:

- elimination of the non-fixed contamination through standard cleaning;
- decontamination using chemical methods or electrochemical procedures (water, acids, bases, oxidants, reducers, etc);
- decontamination using physical procedures (aspiration, abrasion, ultrasound, etc).

For every situation, begin by applying the first method. When it's clear that there is a situation of fixed contamination, then proceed to the second method. The physical method is used only when the chemical decontamination does not provide satisfactory results. The destructive character of physical decontamination is a real disadvantage. Essentially, decontaminated surfaces may be damaged or become unusable because of their tendency to retain contamination easily upon further handling.

The general rule is to decontaminate any object as soon as possible once contamination has been detected. Any contamination which is left to sit for a certain time becomes more and more difficult to eliminate. The effectiveness of decontamination depends on:

- the composition of the radioactive substance;
- the nature of the substrate;
- the amount of time passed between contamination and decontamination;
- decontamination procedures.

Decontamination is always a wet procedure. The procedure is the same no matter the material to be treated, only the cleaning substance changes:

- wash the object in a detergent solution heated to a certain temperature. Non-fixed contamination or contamination attached to an oily substance is easily eliminated this way. Also, using the same solution, lightly rub the object;
- wash the object in water and dry it;

- check for contamination;
- if this first decontamination proves insufficient, repeat the procedure and rub it a bit harder. Leave it to soak for a short time in the detergent solution, or at a certain temperature;
- wash the object in water and dry it;
- check for contamination;
- if the contamination still remains, use other chemical methods which will depend on the extent and the nature of the residual contamination, even on the chemical composition of the substrate;
- if all else fails, prepare to proceed with the appropriate physical decontamination;
- lastly, if this fails, treat the object as radioactive waste.

3.6.4. Personal decontamination

For facial contamination, the procedure is as follows:

- use a gentle soap and a lot of water to thoroughly wash the face (keeping eyes and mouth closed);
- if the eye, ear or mouth has been contaminated, call upon a specialized doctor.

4. Managing radioactive sources

4.1. Recording and labeling

All radioactive sources must be recorded, and the inventory documents must contain the following information:

- radionuclide
- activity
- reference date
- chemical nature of the substance;
- physical state.

These indications will also be displayed along with the radioactivity symbol on the container holding the radioactive substance, on the radioactive source itself if its shape allows, or when necessary, on its container.

If the radioactive substance is ever divided, the data must be updated and the obtained sources must also be recorded.

In certain situations, the supervising authority will ask for an inventory of all radioactive sources in a department or an annual inventory of any radioactive substance purchases.
An inventory may also be requested with respect to giving radioactive sources to a third party or upon disposal of radioactive waste, in order to keep track of all radiation sources.

4.2. Storing radioactive substances

For storage, radioactive sources must be sealed and their containers must be free from contamination. Furthermore, containers holding liquid must be placed into retaining vessels in order to avoid spreading contamination in the event of a spill.

Shielding of stored sources must guarantee that their dose rate at 1 m from their surface remain lower than 0.1 mSv/h.

The storage area (safe, metal locker, chilled locker, freezer) must be set aside for this explicit purpose and distinctly labeled with a warning sign, “ionizing radiation”. This condition limits access to the individuals trained to handle radiation sources.

The nature and the thickness of shielding will be adapted to the type of radiation and the activity of the sources and will ensure the lowest possible rate of ambient radiation. In every situation, the dose rate induced by storing radioactive sources must be lower than 0.1 mSv/week in the places where individuals may remain for a short period of time and lower than 0.02 mSv/week in places where individuals will remain for any extended period of time.

If there is a risk of a gas leak, the storage area must be ventilated.

4.3. Contamination monitoring

Sealed radioactive sources must be periodically monitored for contamination (generally once a year). The goal of this procedure is to verify whether the radioactive substance has escaped its source. Because this procedure may be a source of irradiation, it should be planned and prepared by the radiation protection officer.

4.4. Disposing of sources when they are no longer used

As a general rule, suppliers of radioactive sources must accept the return of any sources no longer used, in order to recycle them or dispose of them as radioactive waste. When it is not possible to return sources to the supplier, the holder of a radioactive source is responsible for its disposal according to current regulations, a procedure which may be costly depending on the type of source and its activity.

5. Transporting radioactive substances

5.1. Transport within the company

Within a given company, ORaP has set conditions for transporting radioactive substances. The following requirements must be met:

- radioactive substances must be packaged in such a way to prevent any leakage
- transported radioactive sources must be subject to constant surveillance;
- the radioactive substance must be labeled with the warning sign, “ionizing radiation”;
- ambient dose rate must be lower than 0.1 mSv/h at 1 m from the surface of the packaging and lower than 2 mSv/h at the surface;
- contamination of the external surface of the packaging must be lower than SC;
- an unbreakable container must be used for all liquid, gas or powder sources;
- for liquids, the packaging must include absorbent material to avoid dispersion in the event of any leakage;
- sources must not be placed in uncontrolled areas (administrative locations, corridor, offices, etc).

The transport of non-packaged radioactive sources may occur, but only under the supervision of a radiation protection officer.

5.2. Current legislation for public transport

Transport regulations consider radioactive substances Class 7 hazardous materials. Any transport beyond the company is thus subject to the following regulations and prescriptions:

SDR	Federal Ordinance concerning the transport of dangerous goods by road.
ADR	European agreement concerning the international carriage of dangerous goods by road.

RID / RSD	International regulation (Swiss) concerning the transport of dangerous goods by train.
IATA	Regulations for transporting regulated articles.

Below a certain activity, called the exemption limit and listed in the ADR, regulation requirements no longer apply.

The ADR distinguishes “special” radioactive sources from other radioactive sources. Special means any form not susceptible to dispersion. Any “special” source must be certified as to its mechanical and thermal resistance and well as its resistance to immersion.

The ADR uses parameters specific to each radionuclide as a limit value:

- exemption limit in specific activity in Bq/g or in absolute activity per package in Bq;
- maximum activity as Type A package for a “special” source (A1);
- maximum activity as Type A package for a non-special source (A2).

5.3. Postal packaging

Requirements concerning the packaging of radioactive materials depend on the activity of the radioactive substance as well as its radiological and dispersion properties. There are 5 types of packaging for which precise requirements have been set:

- special package: the package must be solid, easy to handle and will prevent any dispersion of the radioactive substance, under normal transport conditions. A special package may receive a fraction (10^{-2} to 10^{-4}) of the activity limit A_1 or A_2.
- industrial package: packages must satisfy the generally applied requirements for industrial packaging. There are three types – type 1 (IP1), type 2 (IP2) and Type 3 (IP3). Industrial packages are used for transporting large volumes of material with low specific activity or surface contamination.
- Type A package: beyond the conditions set for industrial packages, Type A packages must include a confinement envelope and be able to resist a series of mechanical stresses corresponding to bad transport conditions. A type A package is what is usually used to transport radioactive sources. It can contain up to activities A_1 or A_2.
- Type B package: the resistance requirements are stricter than those for Type A packages. There are two types – Type B(U) which satisfies all applicable requirements for Type B packages and Type B(M) which do not fulfill all the requirements. Special authorization must be acquired to use a Type B(M) package. Type B packages resist damage which might occur during any land-based accidents. It allows for the transport of activities including between A_1 or A_2 and 3000x A_1 or A_2 via land or sea route.
- Type C package: the resistance requirements are stricter than those for Type B packages. Type C packages are meant to resist the forces which would apply during an air accident. This package is meant for air transport of activities higher than A_1 or A_2, or land or sea transport of activities higher than 3000x A_1 or A_2.

In addition, there are special rules for fissile materials linked to nuclear security.

5.4. Marking and labeling packages

Packages must be marked with the following indications:

- identification of the sender and / or addressee
- UN number, preceded by the indication “UN”

- official name of the transported material (except for special packages)
- package weight (if over 50 kg)
- type of package (IP-1; IP-2; IP-3; A, B(U); B(M); C)
- reference number provided by the supervising authority and serial number of the package (Package type B and C)
- warning sign “Radiation” resistant to fire and water (Package type B and C).

In addition, packages other than special packages containing radioactive substances must come with special labels attached to both lateral sides, carrying the following indications:

- maximum dose rate on the package surface when then defines the label category (I, II or III);
- package contents; must specify the material whose presence poses the main danger in case of damage to the package; (Example: Strontium-90);
- activity; given in multiples of Bq (kBq, MBq, GBq, etc);
- transport index (TI); this number is calculated using the dose rate at 1 m from the package surface. TI is equal to the dose rate measured in mSv/hr, multiplied by 100 and rounded to the first decimal (see Figure 10.3);
- Dangerous material class (7).

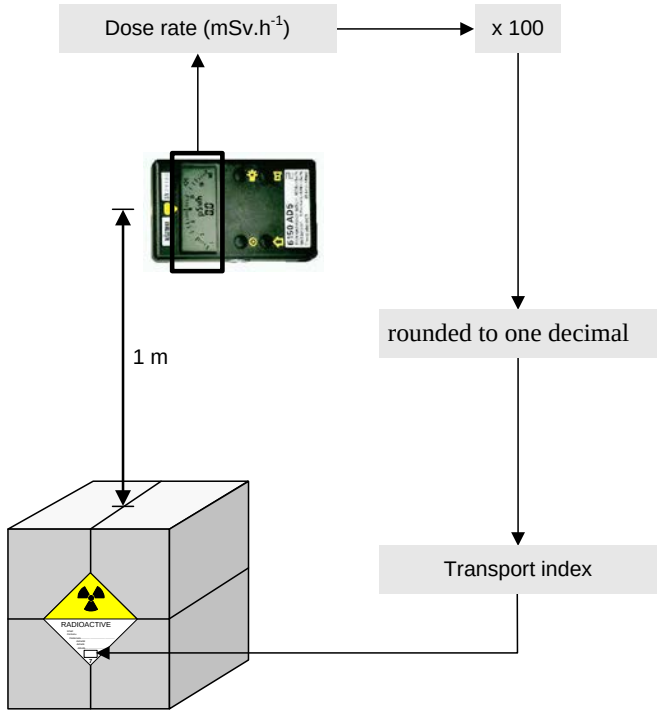
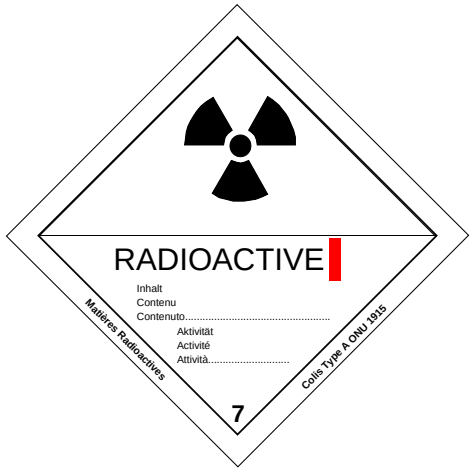


Figure 10.3: Procedure for determining transport index

Warning label – Category I - white

- Maximum dose rate on the surface of the package: 5µSv/h
- Transport index: < 0,05



Warning label – Category II - yellow

- Maximum dose rate on the surface of the package: 0,5 mSv/h
- Maximum transport index: 1,0

Warning label – Category III - yellow

- Maximum dose rate on the surface of the package: 2 mSv/h
- Maximum transport index: 10,0

5.5. Transport documents

Each package containing radioactive materials is accompanied by a transport document which must include:

- the notice: “Merchandise and packaging comply with ADR requirements”;
- UN identification number and name of the material;
- the name and symbol of each radioisotope or the most important isotope;
- the physical and chemical form of the material;

- activity of radioactive materials in Bq or a multiple of Bq;
- package category: I-WHITE / II-YELLOW / III-YELLOW;
- Transport index:
- if necessary, the notice “Fissile material”;
- the identification mark of authorization certificates for special forms, special arrangements, type B or C packages, etc;
- if necessary, the notice “Sent for exclusive use”.

5.6. Safety notice

With the transport document, the sender must establish a safety notice meant for the driver or the intervention organization. It describes:

- load contents;
- dangers related to the transported material;
- protective equipment needed for the transport;
- emergency procedures in case of an accident;
- procedures in case of leakage or spill;
- how to put out a fire;
- precautions needed when giving first aid to any injured person;
- who to inform in case of an accident.

5.7. Vehicle requirements

Vehicles transporting radioactive materials must be equipped with the standard equipment required for transporting dangerous goods, and any equipment needed to take the first measures described in the safety notice. Although not required, a measuring instrument is recommended for monitoring dose rates on the package surface, in the vehicle and in the driver’s area.

When transporting radioactive substances, vehicles must have the following warning signs:

- trefoil with the words “Radioactive” on each lateral side and on the rear wall.
- orange panels on the front and rear.

Any vehicle transporting radioactive substances must have civil liability insurance covering the transport of dangerous goods at a minimum limit of 6 million, and this must be mentioned on the vehicle’s registration. These requirements do not apply to special packages.

5.8. Driver training

Drivers of vehicles transporting dangerous goods are trained regarding the special nature of these transports. Individuals who frequently drive this kind of material receive proper training, confirmed by a certificate. The training certificate is valid for 5 years and must be renewed through a follow-up course before it expires.

There is no special training required if the driver only transports special packages.

5.9. Requirements concerning the company transporting radioactive materials.

5.9.1. Authorization

Transporting radioactive materials falls within applicable law and the ordinance on radiation protection. These two documents require anyone handling radioactive materials to have authorization from the Federal Office of Public Health. By “handling” we mean production, fabrication, treatment, selling, mounting, using, storage, transport, disposal, import, export,

transit and any other form of transfer to a third party. And so the transporter must have an authorization and the sender of the radioactive materials is required to monitor that the company it selects has one as well.

5.9.2. Safety advisor

Each company whose business involves the road transport of hazardous goods or packaging operations, loading, filling or unloading linked to these transports, designates one or several safety advisors who are in charge of preventing risks for individuals, goods or the environment that may be related to these activities.

The safety advisor must hold a professional training certificate valid for road transports. This certificate is awarded by the supervising authority or by the body designated to do this by each contracted party.

To obtain the certificate, the candidate must follow an approved training program and successfully pass a test designated by the supervising authority of the contracted party.

The test is organized by the supervising authority or by an examining body designated by the authority.

5.10. Postal transport

Radioactive materials may be sent through the postal service, as long as the following conditions are met:

- only special packages containing activity lower than $1/5^{th}$ the amount authorized for this type of package can be sent by post;
- the sender must be listed on the package and on all interior labels, if applicable;
- the words “Radioactive” must be displayed inside the package;
- the words “UN no, radioactive materials, special package, Class 7 - ADR - SRF" must be written across the package;
- the transport document and the safety label are not necessary

6. Managing radioactive waste

Handling radioactive substances in the laboratory leads to the production of radioactive waste. By radioactive waste we mean the residue of the radioactive substance, sources which are no longer used, even objects which have been used for decontamination (tissue, cloth) or those which were not able to be decontaminated.

In one way or another, radioactive waste will be released into the environment. We use the decay of radionuclides and their dispersion to make it so that when they eventually re-concentrate, they no longer present any irradiation risk to an individual.

Managing waste begins with production. The ALARA principle also applies to this field of radiation protection and the first goal of a producer of radioactive waste is to reduce the amount of waste to a necessary minimum, lowest activity and volume. To reach this goal, we use only the minimum activity necessary, reduce to a bare minimum the number of objects coming into contact with the radioactive substance and to strictly separate contaminated waste from non-contaminated waste.

6.1. Waste collection

Radioactive waste is eliminated in containers reserved for this explicit purpose. On the other hand, since decay is generally the method used to resolve problems posed by this waste, it is

essential to separate radionuclides with different half-lives upon production. To avoid collecting each nuclide separately, create several categories of nuclides with comparable half-lives:

This means separating waste categories which necessitate further treatment. The following overall categories exist:

- **low activity, non-aggressive solid waste (gloves, decontamination smears, dirty protection paper, etc).** These will be collected in sturdy plastic bags;
- **aggressive solid waste (needles, broken glass, ampoule cutters, etc).** These will be collected in sturdy containers, combustible if possible (thick cardboard tubes, plastic boxes, etc);
- **high activity solid waste** (sealed sources no longer in use, original vials which once contained radioactive substances, etc);
- **low activity water-based liquid waste** (dilution residues, supernate, etc). These will be collected in sturdy plastic buckets, after neutralization if necessary. Trail tubes will be emptied for separating liquid waste from solid waste, if the later treatment of this waste calls for this;
- **high activity liquid waste** (left over parent solutions, perempted products, etc.). These will be saved in their original packaging and then placed in a sturdy, sealed plastic container;
- **organic solutions of liquid waste (**scintillation liquids). These will be collected in unbreakable containers which are resistant to solvents, counting tubes will be emptied if the further treatment of this waste calls for this;
- **biologically unstable waste** (excretions, animal waste, plant waste, etc). These will be collected in sturdy plastic bags and stabilized using a method appropriate for their later treatment (freezing, mummification, desiccation, etc).

Each full waste container is closed and labeled with the following indications:

- name and address of the waste producer;
- radionuclide(s) present in the container;
- approximate activity;
- date the container was closed;
- physical and chemical nature of the waste.

Knowing the activity is necessary for managing the waste and only the waste producer can estimate this amount reliably. Even if this is only an order of magnitude, this information is more precise than any a posteriori measurements of a container whose geometry may be vague.

The waste container is then stored to await decay. The storage facility is subject to the same rules as the storage area of radioactive sources. In principle, it should be reserved explicitly for this use and guarantee radiation protection with respect to the exterior, meaning the walls must ensure the proper shielding of any radiation emitted by the waste and proper ventilation must be able to eliminate any possible gas emissions. Additionally, access to the waste storage area must be restricted to competent individuals in charge of waste management.

6.2. Waste in the environment

6.2.1. Eliminating solid and liquid waste

When the activity of radioactive waste is sufficiently weak, it can be combined with non-radioactive waste. Annex 2 of ORaP defines the activity limit for any waste to be disposed of in this manner, as a function of waste category, and exemption limits LE are fixed per radionuclide in Annex 3. It involves one of the following conditions:

Type of waste	Specific activity	Absolute activity

Solid and liquid waste	A < LE and < 0,1 μSv/h to 10 cm	A < 100 x LE / month and < 0,1 μSv/h to 10 cm
Waste water	A < 1 % of LE [1)]	A < 100 x LE / month

[1)] Average value per week in waste water from the working area

When the half-lives of stored waste are greater than two months, the length of storage can become very long. In these conditions, it is possible to confer radioactive waste to an organization mandated by the Confederation to collect and manage this kind of waste for the long term. A Federal ordinance sets the conditions for managing this waste as well as the corresponding emoluments.

An annual collection campaign is organized by the Federal Office of Public Health and the Paul Scherrer Institute (PSI) is in charge of collecting and treating this waste. For the waste producer, this disposal operation must be coordinated with its supervising authority.

6.2.2. Disposing of laboratory waste water

There are no restrictions regarding the disposal and piping away of water used for decontamination of people or material. However, disposing of radioactive substances into the environment via waste water is, as a rule, forbidden. If there is a holding tank for waste water, the supervised disposal of radionuclides into the environment is allowable. The Federal Office of Public Health has set limits for maximum disposable weekly activity. These activity limits are calculated by taking into consideration various possible contamination sources in such a way that in no way does the waste water network collect concentrations leading to doses higher than the maximum allowable dose for the public. As a result, any procedure to empty a holding tank must only occur after sampling and analysis of activity.

The supervising authorities generally require the installation of these types of tanks for Type B working areas or when several Type C working areas are linked by the same pipe network. Maximum disposable activities are calculated on a case per case basis by the Federal Office of Public Health and set in the authorization for using radioactive substances. This calculation is based on the specific building's water consumption, which guarantees the dilution of the disposed activity.

6.2.3. Evacuating air from laboratories

For Level C working areas, there is no particular requirements set concerning the evacuation of air from the laboratory. However, the ventilation facilities of Level B labs must come equipped with filters suitable for the radionuclide handled. In these conditions, air contamination becomes negligible. Used filters are generally contaminated and must be considered radioactive waste. For Type A working areas or for certain special situations, the supervising authorities can set a maximum allowable air contamination and require the installation of a continuous control monitor for evacuated air contamination. Maximum disposable activity limit is calculated by taking into consideration various possible contamination sources in such a way that in no way does the air treatment network collect concentrations leading to doses higher than the maximum allowable dose for the public.

7. Summary:

- *Operational parameters are secondary limit parameters; they are routinely measurable and represent an approximation of the primary limit parameters (unable to be measured).*
- *Secondary activity limits are:*
  - *exemption limit, LE*
  - *authorization limit, LA*
- *Guideline values are:*
  - *guideline value for air contamination, AC*
  - *guideline value for surface contamination, SC.*
- *The four rules of protection against external radiation are:*
  - *pre-selection of the least penetrating radiation possible;*
  - *selection of the weakest possible source intensity;*
  - *maintaining the greatest possible distance between the source and the individual;*
  - *reducing exposure time to the strictest minimum;*
  - *Using protective screens*
- *Shielding used must be suitable for the radiation in question;*
  - *for α radiation, the risk of external irradiation is zero because the radiation is stopped in the uppermost layer of the epidermis (dead skin cells);*
  - *for β radiation, a low Z shielding material with an approximately 1 cm thickness ensures effective protection;*
  - *for γ or X radiation, attenuation is exponential, so there is no complete protection. The transmission factor of the screen, which depends on the energy of the radiation and the material of the screen, can be taken from tables or graphs.*
- *Frequent monitoring of the ambient radiation is advised for areas where intense sources are used.*
- *The basic rules for protection against intake are:*
  - *pre-selection of a source having the weakest toxicity and the lowest activity;*
  - *confining the radioactive substance;*
  - *preventing all contact between the individual and the radioactive substance.*
- *Protection is ensured by structures, by personal protection means and appropriate working methods.*
- *The type of working area is determined by the authorization limit (LA) of the radioelement and by the activity; we distinguish, by increasing radiological risk and protection level: normal laboratory, Type C working area, Type B working area and Type A working area.*
- *The requirements concerning the different types of working areas are established in the technical ordinance from the Federal Department of the Interior.*
- *There is a difference between direct contamination measures and those conducted using a smear test.*

Decontaminating surfaces and objects is done using mechanical, chemical and

physical methods.

Skin decontamination is carried out by washing with a mild soap (pay attention not to break the skin).

- *Maximum residual contamination corresponds to SC outside controlled areas and to 10 x SC inside controlled areas.*
- *Any radioactive source must be labeled and recorded with the following indications:*
  - *radionuclide*
  - *activity and reference date;*
  - *physical and chemical nature.*

  Radioactive sources must be stored in shielding so that the dose rate at 1 m from their surface is below 0.1 mSv/h.

  They are then stored in marked areas, set aside for this purpose. These areas must be shielded so that the dose rate outside the storage area is lower than 0.02 mSv/week.
- *Transporting radioactive sources within a company is supervised by ORaP, while transport in the public domain is covered by the ordinance on transporting dangerous goods (SDR) which must be respected. Postal regulations define the conditions under which dangerous goods may be sent through the post.*

  There is a difference between special sources, accompanied by a certificate guaranteeing their mechanical and thermal resistance, and other sources.

  The maximum activity that can be put in a Type A package is given by the values A_1 or A_2 which figure in the SDR. Above A_1 or A_2, a type B package is required.

  An activity on the order of 10^{-2} to 10^{-4} A_1 or A_2, depending on the radioactive source, may be transported in a special package (1/5 of this activity for postal packages).

  Type A or B packages must carry warning labels.

  Radioactive packages must be accompanied by a transport document and a safety notice in case of accident.
- *Radioactive waste must be separated at production according to characteristic (half-life, activity level, physical state, chemical nature, safety risk, etc). They are then stored for decay.*

  They can be disposed of into the environment when their activity is below the LE in specific activity or 100 x LE/month in absolute activity.

  Waste having a half-life longer than two months can be disposed once a year during a collection campaign organized by the FOPH.

  Laboratories using holding tanks for waste water may obtain authorization to eliminate a certain weekly activity through waste water.

  There are no restrictions on disposing of decontamination water.

  Air evacuated from B laboratories must be filtered upon exit using ventilators and filters are then treated as radioactive waste.

8. Practice questions

1. Calculate the committed effective dose received by a person who inhales an activity equal to the LA of Iodine-125. Indicate as well the dose equivalent to the thyroid as if this was the only organ irradiated.

2. Estimate the annual dose for a person working 10 h per week at 1 m from a 500 kBq source of Cesium-137 (control source).

3. Calculate the annual surface dose for a person contaminated with Strontium-90 equal to the SC value.

4. Compare at equal transmission (1%) the thickness of lead to be used for an area where a source of Cobalt-60 or Iridium-192 is used.

5. Calculate, for a surface dose, the benefit linked to using tweezers (handle length; 20 cm) to hold a flask of Technetium-99m.

6. If the operation using the tweezers in Exercise 5 lasts 1 minute, Calculate the duration of the manual manipulation if we want to maintain the same dose.

7. Calculate the thickness of the wall (concrete) of an area (25 m2) where a 20 GBq source of Cesium-137 is used for 10 h per week, and knowing that the adjoining area is occupied by public individuals (0.02 mSv per week).

8. Determine the required working area for handling an open source of 10 MBq Strontium-90.

9. Indicate the effective dose received in case of inhalation of an activity of Strontium-90 corresponding to the authorization limit.

10. Calculate the waiting time so that a surface contamination of 200 Bq/cm^{-2} of Iodine-125 decays below the legal guideline value for uncontrolled areas.

11. What kind of package must be used for a 400 MBq source of ^{241}Am?

12. What kind of warning label must be affixed to the package from Exercise 8 if no radiation shielding is planned?

13. What is the transport index for a package whose dose rate at 1 m from the surface is 20 μSv/h ?

14. How long must solid waste from a 50 MBq ^{32}P source be stored before being able to throw it in the garbage?

15. How do you dispose of 1 GBq of liquid ^{125}I?

CHAPTER • 4

PERSONAL RADIATION PROTECTION MONITORING

Course goals
• *Explain the methods of individualized monitoring for external irradiation and intake.*
• *Calculate doses delivered in an intake situation.*
• *Cite and describe the function of a dosimetric document.*
• *Design dosimetric monitoring in special situations*

1. Role of personal monitoring in radiation protection

Personal monitoring involves all methods used for determining an accumulated dose for an individual. This accumulation is used for determining:

- the risk incurred by the individual. Frequent dose measurements are used to ensure that the dose remains as small as possible and that it remains lower than the maximum legal limit in any situation. If accumulated dose approaches or exceeds allowable limits, a re-examination of working conditions should be planned in order to reduce personal doses and ensure the safety of the worker.
- risk associated with a particular job. Personal monitoring will highlight all tasks and working locations where irradiation is the greatest. These observations will ensure the strengthening of protective measures where they will be the most useful.
- professional errors. An unjustified dose increase through any kind of work modification may lead to an accident. This situation will be thoroughly examined in order to avoid being repeated. Tasks performed often by several laboratories or companies. By comparing respective doses, it is possible to discover qualities and errors (from a radiation protection standpoint) regarding the methods employed;
- risks involving the overall population. By summing personal dose, we can calculate the contribution of different professions to the irradiation of the general population, thus facilitating a cost-benefit analysis of each activity.

The goal of medical testing is to verify the aptitude of the person working with ionizing radiation and ensure the appropriate follow-up of his or her health.

2. Personal monitoring of external irradiation

2.1. Parameters monitored

Parameters involved in monitoring measurements are the operational parameters of external radiation. The following:

- personal deep dose $H_p(10)$ (abbreviated to H_p); this is the dose equivalent at 10 mm deep into soft tissue at the thorax; measuring this parameter is done using a dosimeter worn on the chest

(badge); in general, the dosimeter's sensitive element is covered with a filter to simulate absorption at 10 mm into tissue;

- personal surface dose $H_p(0,07)$ (abbreviated to H_s); this is the dose equivalent at 0.07 mm deep into soft tissue at the thorax; measuring this parameter is done using a dosimeter worn on the chest; for this measurement, the dosimeter's sensitive element is generally directly placed into the radiation field (no filter);
- dose equivalent to the extremities, H_{ext}; dosimeters are placed near the hands (bracelet, rings, etc).

When the measured dose H_p is lower than the limit value of the effective dose, we accept H_p as the dose received (registered in the dosimetric document). Similarly, when the measured dose H_s is lower than the limit value of the surface dose, we accept H_s as the surface dose received (registered in the dosimetric document). In a situation where the measured doses are higher than the limit values, a test is done, on the basis of which the effective dose and the surface dose are determined using dosimeter readings, and taking into consideration the specifics of the irradiation.

2.2. Measurement technique

Measuring instruments include:

- integrating dosimeters carried on the chest (badges);
- integrating dosimeters worn near the hands (rings);
- direct read or alarm dosimeters.

2.2.1. Badge dosimeter

The measurement technique depends on the type of radiation. For β, γ or X-ray radiation, the most common detection systems include thermoluminescence (see Figure 4.1.a), photographic film and radiophotoluminescence.

The following requirements are set:

- minimum detectable dose : 0,1 mSv
- maximum detectable dose : 5 Sv
- variation of response with γ energy : < ± 40 %
- variation of the γ response with direction : < ± 40 %
- reproducibility (2 σ) : < ± 20 %
- variation of the response caused by other effects (fading, humidity) : < ± 10 %.

A dosimeter responding to these requirements should make it possible, under routine conditions, to measure dose with a +50% to -30% precision as set in the ordinance on personal dosimetry.

Dosimetry services should have official approval and participate once a year in a dosimetric intercomparison.

Monthly measurements are required. However, in the event of an accident or if any irradiation is suspected, an immediate reading of the dosimeter is possible.

When wearing a protective apron, the dosimeter should be worn under the apron. If you suspect that any unprotected part of the body might be exposed to significant irradiation, it is advised to wear a dosimeter on the unprotected part of the body or over the apron.

Dose is calculated as follows when wearing two dosimeters:

$Hp = Hp_{under} + a \cdot Hp_{over}$ with a = 0.1 if the apron does not protect the thyroid gland

$Hs = Hs_{under} + Hs_{over}$ a=0.05 if the apron protects the thyroid gland

where H_{under} and H_{over} represent the dose measured under or over the apron, respectively. This signifies that we allow that 90 to 95 % of all risk is located near the organs protected by the apron.

2.2.2. Ring dosimeter

This kind of detector is generally a thermoluminescent dosimeter placed inside a ring, which measures the dose received by the hands. In certain handling situations, we place the ring at the end of the fingers to measure contact dose.

(a) 

(b) 

Figure 4.1: Thermoluminescent dosimeter (b) electronic dosimeter with alarm

2.2.3. Direct read dosimeters or alarm.

Every time the risk involved is high or poorly understood, the badge is combined with a direct-read instrument (see Figure 4.1.b). The pen dosimeter, based on the discharge from a condenser, gives an immediate reading. The alarm dosimeter, based in general on a GM counter, a proportional counter or a semi-conductor detector, includes an electronic measuring device which can calculate dose. It generally delivers an intermittent or continuous alarm if a dose threshold or dose rate has been exceeded.

Currently, electronic dosimeters are used only in high dose rate situations. However, much research has been conducted on the international level to develop electronic dosimetry for personal monitoring, since this approach provides several advantages for improved monitoring of doses and facilitates the effective implementation of optimization programs, notably:

- good precision, since there are less variables related to reading;
- good exactitude, thanks to improved precision and long-term stability;
- good detection limit, meaning greater reliability at lower doses;
- visual and audible alarms, dose and dose rate function;
- easier reading, which means a reduction in dosimetric services;
- easy transfer of dosimetric information to a data base, which facilitates and speeds up data processing;
- good acceptability with users who have more confidence in a dose value they receive immediately.

However, currently, the introduction of electronic dosimeters for routine personal monitoring is slowed down by several remaining problems, notably;

- absence of national or international industrial norms (standards)
- the reticence of monitoring bodies toward these new dosimeters;
- relatively high price;
- underestimation of dose at high rates (some Gy/h), caused by the effects of dead time, which poses a problem in an accident situation;

- electromagnetic interference;
- difficulty in measuring surface dose;
- lack of dosimeter for extremities.

3. Personal monitoring for external contamination

Frequently, when working with open sources, and at least when leaving the laboratory, everyone is required to monitor skin and clothing contamination. The instrument used is not individual, but everyone takes a turn passing in front of the counter, taking care to present their hands, feet and then their entire body.

If a positive result is obtained (value higher than CS), it is necessary to clean oneself or change clothing until activity is reduced below the tolerated threshold. Doses, which may be delivered in this way, are difficult to estimate and quite low. They are not calculated, nor taken into consideration.

4. Personal monitoring for internal contamination

4.1. Introduction

Incorporating a radioactive substance leads to internal irradiation. Determining doses linked to this kind of situation is relatively complicated, because it involves not only the physical parameters of the incorporated radionuclide, but also the chemical characteristics of the radioactive substance which determines how the product is metabolized in the body.

In practice, we define a certain number of derived limits which allow for a simple way of verifying that personal dose limits have been respected.

Measuring an intake is done in several different ways, either through determining activity in the entire organism or in an organ, or measuring activity in the feces. Interpreting these measurements is done by comparing to secondary limits.

4.2. Determining dose linked to intake

4.2.1. Calculating the dose to an organ

Consider the intake of a unit activity of a radionuclide. The substance spreads out through the organism depending on its metabolism and is eliminated after a short or long time period. When decay occurs in the source organ s, the dose in the target organ t, produced by type i radiation is called specific effective energy and is symbolized as follows:

$S(s \to t)_i$

This parameter, which is one of the basic elements of the dosimetry of intakes, depends only on the energy of the i radiation and the spatial relationship between the s and t organs.

Total dose linked to decay in organ s is obtained by summing all the particles emitted during decay:

$$H_{50}(s \to t) = U_S \cdot \Sigma_i \, S(s \to t)_i = U_S \cdot S(s \to t).$$

U_s being the number of decays which occur in the source organ s per unit of activity incorporated during a period of 50 years.

In order to obtain the total dose to the organ t, you must sum up the contributions of all the source organs s and in the presence of a radioactive decay chain, all the nuclides formed j. We get:

$$H_{50,t} = \Sigma_s \Sigma_j U_s \, S(s \to t) \cdot$$

This formula shows that by knowing the parameters U_s and S, the dose is obtained through simple summation.

And so for an inhalation situation:

$$E_{inh} = I \cdot \Sigma_T w_T \Sigma_s U_{s,inh} \, S(s \to t) = I \cdot e_{inh} \, .$$

Where I is the incorporated activity, and the complex summation, symbolized by e_{inh}, is the effective dose linked to the inhalation of an activity of 1 Bq. This parameter is indicated for all radioelements in Annex 3 of ORaP.

The situation is the same for ingestion:

$$E_{ing} = I \cdot \Sigma_T w_T \Sigma_s U_{s,ing} \, S(s \to t) = I \cdot e_{ing} \cdot$$

4.2.2. Number of decays in a source organ for 50 years

Metabolizing substances in the body is a complex process (see Figure 4.2). For dosimetric purposes, we use a compartmentalized model with first order kinetics.

After inhalation or ingestion, the radionuclide passes into the extracellular fluid which is simulated by a transfer compartment uniformly divided across the entire body. Its biological half-life is set at 0.25 days. The radionuclide then passes into the various organs before being eliminated with body waste.

Special models were developed by the ICRP for the respiratory system, the gastro-intestinal tract, bone retention and the respiration of noble gases.

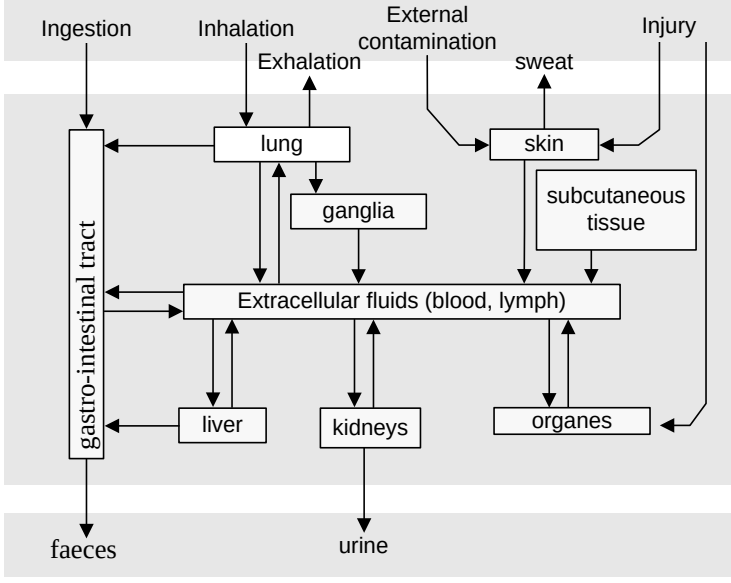


Figure 4.2:Principal metabolic pathways for radionuclides in the body

4.3. Measurement techniques for internal contamination

4.3.1. In vivo measurements

Once a radioisotope has been absorbed into the body it can emit γ radiation with enough energy to be detected from outside the body. A counter, placed near the individual can measure emitted radiation. The most common measuring situations are the following:

- measuring radiation emitted by the entire body. The patient and detectors are placed inside a shielded area in order to prevent the detectors from detecting natural irradiation. This is a necessary step since acceptable concentration levels for the body are very low for many isotopes and difficult to measure. This relatively cumbersome technique works well for detecting elements which disperse themselves more or less uniformly throughout the body, K, Co, Na… for example;
- measuring radiation emitted by the thyroid gland. This is done mainly for radioactive iodine intakes which concentrate for the most part (20 to 40%) in the thyroid; it is therefore easy to measure with a shielded probe placed adjacent to the thyroid gland. Measuring time takes approximately a half an hour. The instrument used is a scintillation detector, generally a NaI crystal which makes it possible to identify radioelements through spectrometric analysis.

4.3.2. In vitro measurements

Analyzing radioactivity of a biological sample makes it possible to determine internal contamination. Although urine is typically used because it is easy to sample, and also because it tends to be representative of body waste, there are situations in which measuring the activity of blood or stool samples is more appropriate.

When γ emitters are involved, the measuring technique used on a sample is similar to the one used for in vivo measurements. However, it is difficult to attain good sensitivity with this method. When β or α emitters are involved, the sample is measured using liquid scintillation or by counting after preparation on a thin layer.

4.4. Determining dose after intake

4.4.1. General information

In an intake situation, the dose received depends on many factors:

- the quantity of incorporated radionuclide;
- the type and energy of the emitted radiation;
- the half-life of the radionuclide;
- the intake mode (inhalation, ingestion, contamination from a wound),
- metabolism of the incorporated substance.

In an intake situation, the operational parameter to determine is the committed effective dose E_{50}. When we know the incorporated activity I, the committed effective dose is calculated as follows:

- for ingestion : $E_{50} = I.e_{ing}$
- for inhalation : $E_{50} = I.e_{inh}$.

Monitoring intake, the goal of which is to measure intake, I, and through that the committed effective dose E_{50}, is based on:

- for penetrating radiation (γ), using the radiation emitted by the individual (for example using a thyroid monitor or a whole body counter);
- for low penetrating radiation (α, β), using the measurement of body waste (urine, feces) contamination.

Individuals working regularly with open radioactive sources should be monitored for intake when the intake of the nuclide in question could lead to an annual effective dose greater than 0.1 mSv.

4.4.2. Measuring procedure

There are two types of procedures:

- measuring the intake itself; this is a measurement used to determine E_{50}; it should be conducted by an authorized dosimetry service;
- screening measurement; this is a simple measurement which ensures that the incorporated activity remains below a threshold; this threshold is usually set so that an individual will receive an annual dose lower than 1 mSv. Figure 4.3 gives the decision diagram for monitoring intake using a screening measurement.

4.4.3. Measurement intervals

Measurement intervals will depend on the effective half-life of the radioelement and the detection limit. In practice, we consider that the intake occurred in the interval between 2 measurements. We select the measurement interval so that the intake at the beginning or the end of the interval does not lead to an underestimation or an overestimation of a factor 3. There are, however, situations where this requirement cannot be fulfilled because detection limits are too high (actinide intake).

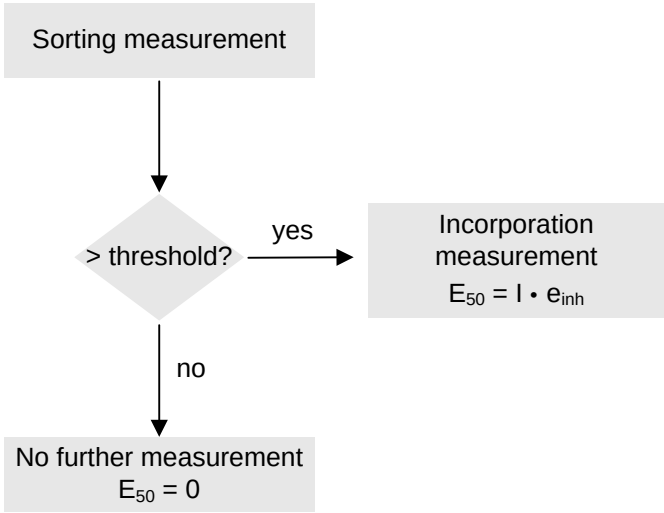


Figure 4.3: Decision diagram for monitoring intake using a screening measurement.

4.4.4. Calculating committed effective dose

To calculate the committed dose, consider that the activity was inhaled. The parameter m(t) represents the fraction of the inhaled activity (I) located in the organ or waste sample at time t. It has been determined for standard conditions (average metabolism) in the literature (publication 54 of the ICRP). For example, Figure 4.4 indicates this parameter for the activity in the thyroid after an inhalation of Iodine-125.

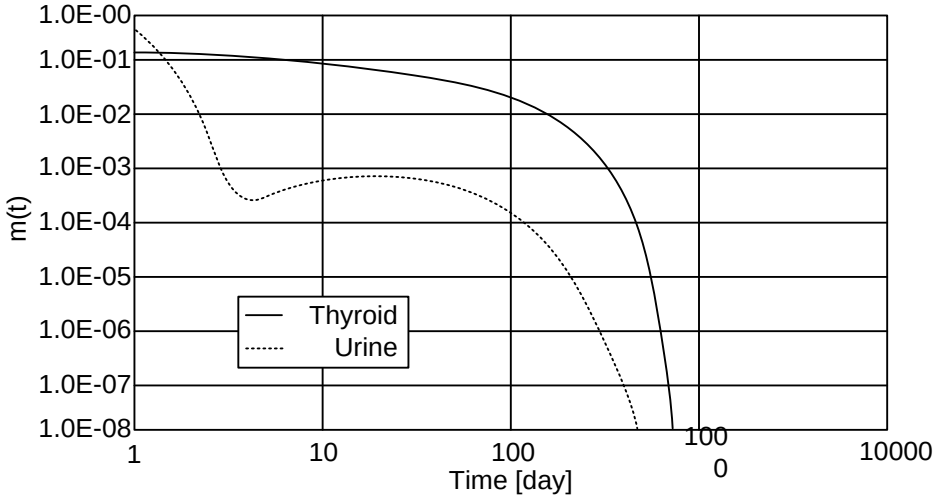


Figure 4.4: Retention of Iodine-125 in the thyroid after inhalation

If we measure an activity M(t) in the organ or body waste at time t, we can calculate I as follows:

$$I = \frac{M(t)}{m(t)}$$

and the same for E_{50}:

$$E_{50} = I \cdot e_{inh} = \frac{M(t)}{m(t)} \cdot e_{inh} = M(t) \cdot \frac{e_{inh}}{m(t)}$$

.

For routine monitoring, when we consider that the intake occurred in the middle of interval T, the committed effective dose is written:

$$E_{50} = M(t) \cdot \frac{e_{inh}}{m(T/2)}$$

.

In a situation where a part of the intake detected in the previous measurement is still present in the organ, the formula becomes:

$$E_{50} = M(t) \cdot \frac{e_{inh}}{m(T/2)} - E_{50}^{a} \cdot \frac{m(3T/2)}{m(T/2)}$$

where E_{50}^{a} is the committed effective dose measured during the previous control.

4.4.5. Measures to take if the intake exceeds dose limit

When the dose is determined with the standard method described above and exceeds the limit value, a specific determination of the dose must be conducted, taking the specific situation into consideration:

- supposed moment of intake;
- individual's specific metabolism;
- specific intake path;
- chemical properties of the radioelement.

This determination is given to a radiation protection officer who will conduct an inquiry as well as additional measurements with the individual.

4.4.6. Special situations

There are situations in which the standard methods do not apply, for example intake uniquely through ingestion, special chemical form, particular physical state (dimension of the aerosol).

In this case, other models need to be applied after approval of the supervising authority.

4.4.7. Parameter values for calculating intake

Parameter values needed for monitoring and calculating engaged dose:

- monitoring interval,
- measurement threshold,
- measuring method,
- $e_{inh}/m(t)$ value,

are subject to a recommendation from the Swiss Federal Commission on Radiation Protection. They are integrated into the Ordinance on Dosimetry.

5. Medical monitoring

A medical exam is conducted;

- at the beginning of the activity,
- periodically, according to a schedule defined by the Swiss Accident Insurance Fund (SUVA) (usually every two years for situations involving a high risk of irradiation; when risk is low, periodic monitoring is advised).

The exam involves a complete blood work up.

However, it must be specified that the goal of the medical exam is not to observe symptoms linked to irradiation since only deterministic effects are detectable. The medical exam is part of the approach for preventing professional illnesses defined by the law on accidents, a task which falls under the responsibility of the SUVA.

6. Personal dosimetric document

Each individual with occupational exposure to radiation has a personal dosimetric document, on which his/her received doses are indicated. This document is updated by the employer and given to the individual when they leave their place of employment. If the individual is re-engaged with another job involving exposure to radiation, the dosimetric document is given to the new employer.

7. Summary:

- *Personal monitoring of external radiation is done using a personal dosimeter worn on the chest (badge). It measures the parameters H_p(personal deep dose) and H_s(personal surface dose).*
- *The sensitivity of the dosimeter is 0.1 mSv.*
  Its precision is generally from – 30% to +50%.
- *When wearing a protective apron, the dosimeter should be worn under the apron.*
- *When there is a particular risk of irradiation to the hands, a ring-dosimeter should be worn.*
- *A dosimeter with an alarm is used, in addition to the personal dosimeter, when the risk involved is high or the irradiation situation poorly known.*
- *Intake measurement is done:*
  - *either in vivo; for penetrating radiation (anthropogammametry, thyroid monitor);*
  - *or in vitro; for low penetrating radiation (tritium measurement and Carbon-14 in urine, measure of actinides in a stool sample).*
- *The dose linked to an intake is expressed as follows:*

$$E = I \cdot \Sigma_T w_T \Sigma_S U_S \cdot S(s \to t) = I \cdot e$$

It is the product of the following three parameters:

- *I; the incorporated activity;*
- *$S(s \to t)$; the dose to the target organ t through decay in the source organ s ; this parameter is a function of the type of particles emitted and their energies, as well as morphology;*
- *U_s; the number of decays in the source organ s per unit of activity incorporated.*

- *There are two types of intake situations: inhalation (through respiration) and ingestion (orally). Therefore:*

$$E_{inh} = I \cdot e_{inh} \quad \text{et} \quad E_{ing} = I \cdot e_{ing} .$$

The parameters e_{inh} and e_{ing} are listed in Annex 3 of the Ordinance.

- *To calculate an intake:*
  - *we consider it as an inhalation;*
  - *we set the measurement interval to ensure precision to a factor 3 (approximately 3 effective half-lives);*
  - *we consider that the intake occurred between two intervals;*
  - *we proceed with a screening measurement (yes/no response);*
  - *in case the threshold is exceeded, we conduct an intake measurement;*
  - *dose is calculated using the formula:*

$$E = M \cdot \frac{e_{inh}}{m(T/2)}$$

Where M is the measurement value and m(t) the fraction fixed (or excreted) at time t after intake.

- *Individuals having occupational exposure to radiation are subject to medical monitoring.*
- *Doses received by individuals having occupational exposure to radiation are recorded in a personal dosimetric document updated by the employer.*

8. Practice questions

1. Indicate the dose to record on the dosimetric document if two dosimeters (one under the apron and the other over it) give the following values:

$$H_{under} = 0,4 \text{ mSv} \quad ; \quad H_{over} = 2 \text{ mSv}.$$

2. Indicate a monitoring program for an individual working with an open source of Tritium.

3. We measure a radioactive concentration of 10^6 Bq.1^{-1} of Tritium in the urine of a worker during a screening measurement. What should we do ?

4. Calculate the committed effective dose received by an individual who has an Iodine-125 activity of 2 MBq in the thyroid gland during a quarterly exam.

5. We measure an activity of 5 MBq of Cobalt-60 during an anthropogammametric exam. Estimate the dose if we say the intake occurred three months previously.

6. Imagine a monitoring measurement for the intake of Technetium-99m.

CHAPTER • 5

THE ROLE OF THE RADIATION PROTECTION OFFICER

Course goals
• *Understand the legal foundations of the job of radiation protection officer (RPO).*
• *Have a general understanding of the RPO's duties.*
• *Understand personnel training requirements.*

The idea of an officer is set in Article 16 of the law on Radiation Protection. His or her duties must be set in writing by the authorization holder who takes on radiation protection responsibility in his/her company (ORaP, Art. 132). The radiation protection officer (RPO) supports and advises the authorization holder for all questions relating to radiation protection. For all practical radiation protection applications, he/she must be competent and have the necessary means at his/her disposal. The RPO may delegate some of his/her duties. In a company, everyone is involved in radiation protection (director, authorization holder, RPO, employees).

The following tasks relate specifically to the RPO:

- outfitting (and also planning) of working areas
- organizing radiation protection and managing working areas
- monitoring and supervision of working areas and working methods
- managing administrative tasks
- communication with the supervising authority
- basic training and continuing education of collaborators in radiation protection practices.

1. Outfitting

In terms of outfitting the working areas, the RPO'S duties are the following:

- designation of working areas
- organizing working areas, such as: distribution, outfitting, shielding
- establishing effective working methods from a radiation protection perspective
- acquisition and maintenance of radiation protection measuring instruments;
- acquisition of protective gear (apron, thyroid shields, gloves, white coat…). The RPO verifies that protective gear is available, in sufficient quantities, and is correctly and systematically used.
- preparation of internal guidelines with respect to radiation protection, as well as measures to take in case of accident or fire. The RPO ensures that these instructions are known and applied by the individuals involved.

2. Organization and management

In terms of organizing and managing radiation protection, the RPO has the following duties:

- designate those individuals having occupational exposure to ionizing radiation
- organize personal monitoring. This involves, on the one hand, ensuring that everyone exposed to external radiation wears a dosimeter, and, on the other hand, defining any internal dosimetry needs and establishing necessary screening measurements for internal contamination
- declare to the Suva anyone having occupational exposure to ionizing radiation to ensure medical supervision
- organize and manage purchasing, transport, receiving, storage and disposal of radioactive substances
- manage radioactive waste
- manage laboratory waste water
- organize maintenance and monitoring of installations.

3. Monitoring and supervision

In terms of monitoring and supervising working areas and working methods, the duties of the RPO are the following:

- analyze results of personal dosimetry from individuals with occupational exposure to ionizing radiation and remain in regular communication with those individuals regarding those results
- monitor installations and working areas for contamination and external irradiation
- check shielding and dose rates
- monitor the integrity of sealed radioactive sources
- supervise trials or any work involving any special risks
- regular monitoring in working sectors, mainly in the laboratories
- monitor the stability of installations
- supervise the behavior of individuals without occupational exposure to ionizing radiation (reception, repair services, visitors, etc).

In the medical field, the RPO is in charge of ensuring that protection measures for patients are applied in an optimal manner. This also involves informing patients and their families both before and after a radiological exam, particularly for an exam or treatment involving nuclear medicine.

4. Administration

In terms of radiation protection administration, the RPO has the following duties:

- provide information and internal training for individuals having occupational exposure to ionizing radiation
- update paperwork concerning the acquisition, use and elimination or disposal of radioactive substances
- manage authorizations for using ionizing radiation
- update personal dosimetry documents

5. Communication with the supervising authority

The Swiss Federal Office of Public Health (FOPH), the Swiss Accident Insurance Fund (SUVA) and the Swiss Federal Nuclear Safety Inspectorate (ENSI) are the three Swiss organizations involved in monitoring the protection of individuals and their surroundings. The FOPH monitors the companies in which it is necessary to protect the public, notably medical companies and research institutions; the SUVA is mainly interested in protecting workers by supervising industrial and artisanal companies, while the ENSI primarily takes care of monitoring nuclear facilities.

The RPO must immediately contact the supervising authority in the following situations:

- change in authorization conditions (changes concerning the installation, data involved with the building and the construction of the installation or even the area where radioactive sources are stored)
- purchasing and use of new radiological installations
- exceeding any dose limit values
- radiological incident or accident
- clinical trials with radiation
- suppression of working sectors (stop of activity)
- change of RPO.

Requests to change the conditions of an authorization must be made prior to any change and modifications must not occur until authorization has been received.

6. Training in radiation protection

6.1. Goal of training

The training of individuals who may be exposed to radiation is mandated by Article 6 of the law on radiation protection and the training methods are described in Articles 11 through 22 of ORaP. Details of this training are established in a departmental technical ordinance. Training targets the following objectives:

- acquiring the necessary basic knowledge for understanding the risks associated with radiation and the means of protection
- acquiring the basic principles of radiation protection and practical methods destined to protect workers, patients, the general public and the environment
- acquiring knowledge of the legislation and administrative procedures linked to using ionizing radiation

Since radiation protection is not a science but a technique, training must be essentially practical in nature.

6.2. Training methods

There are several training methods used in radiation protection:

- training integrated into professional training (Medical Radiology Technician, medical lab workers, medical aides, etc)
- training provided within the framework of continuing education specific to the profession (radiologists, nuclear medicine specialists, intervention services, etc)

- training provided within the framework of independent courses (RPOs, lab workers, etc)
- training inside a company (laboratory personnel, technical services, cleaning personal).

6.3. Training level

The following training levels are established:

- radiation protection officers (RPO). Training must be broad and cover the entire field of radiation protection. The officer must be able to work independently, organize personnel and take useful decisions in case of a radiation protection problem. The individual named as officer must prove to have an in-depth knowledge of radiation protection legislation and the tasks of radiation protection specific to their working field.
- collaborator with active participation. Training must allow for an understanding of basic principles, a mastery of problems specifically related to the planned application and appropriate behavior in the event of any radiation protection problem.
- collaborator without active participation. Training must give general information on the planned application and allow for appropriate behavior with respect to ionizing radiation.

6.4. Training centers

Training is provided by:

- the supervising authorities (FOPH, SUVA)
- training centers recognized by the supervising authorities (PSI, Safpro, IRA)
- university or professional training schools whose curriculum has been accepted by the supervising authority and whose final radiation protection exam is supervised by an officer designated by the authority
- the companies and laboratories having a trained RPO, with permission from the supervising authority.

7. Summary:

- *The idea of an officer is established in Article 16 of the law on Radiation Protection.*
- *His or her duties must be set in writing by the authorization holder who takes on radiation protection responsibility in his/her company.*
- *The radiation protection officer (RPO's) main tasks are:*
  - *identifying radiation protection problems*
  - *outfitting working sectors*
  - *organizing radiation protection and managing laboratories*
  - *monitoring working areas and supervising working methods*
  - *administrative tasks, communicating with the supervising authority*
  - *training those individuals with occupational exposure to ionizing radiation*
- *Radiation protection training is defined in the Ordinance on Radiation Protection (ORaP). Details are fixed in a technical Ordinance.*

8. Practice questions

1. Why must the duties of the RPO figure in a written job description?

2. What types of methods does the RPO have at his/her disposal for ensuring application of radiation protection rules in his/her company?

3. How can you ensure the adequate radiation protection training of foreign PhD students coming to complete their dissertation in your department?

4. What is the minimum training needed for an individual working as a laboratory cleaner?

5. Prepare a plan for providing initial information on radiation protection to give to new collaborators coming to work in your department.

6. Who can you ask for information on training opportunities in radiation protection specific to a particular application?

CHAPTER • 6

RADIATION PROTECTION LEGISLATION

Course goals
• *Understand the international references used as a basis for Swiss regulations.*
• *Be able to explain how radiation protection is handled in Switzerland.*
• *Understand the guiding principles of Swiss legislation.*
• *Be able to cite the various technical ordinances linked to radiation protection.*
• *Be able to use the ORaP annexes.*

1. Overall situation

At the international level, radiation protection lies at the heart of several organizations. First, there is the International Commission on Radiological Protection (ICRP). This is a group of specialists from all nationalities who come together from various backgrounds and who are concerned with human exposure to ionizing radiation. The commission, organized into working groups, regularly publishes recommendations in the field of radiation protection. The ICRP's positions are rarely called into question and generally serve as a basis for international regulations or national legislation. ICRP publications also deal with the general field of radiation protection (for example):

- *ICRP Publication 38 "Radionuclides transformations : energy and intensity of emissions",*
- *ICRP Publication 60 "1990 recommendations of the ICRP",*
- *ICRP Publication 89 "Basic anatomical and physiological data for use in radiological protection : Reference values ",*

ICRP Publication 103 " Recommendations of the ICRP for worker exposure ":

- *ICRP Publication 47 "Radiation protection of workers in mines",*
- *ICRP Publication 75 "General principles for the radiation protection of workers",*
- *ICRP Publication 78 "Individual monitoring for internal exposure of workers",*

for patient exposure:

- *ICRP Publication 34 "Protection of the patient in diagnostic radiology",*
- *ICRP Publication 86 "Prevention of accidents to patients undergoing radiation therapy",*
- *ICRP Publication 93 "Managing patient dose in digital radiology",*

for general public exposure:

- *ICRP Publication 29 "Radionuclides release into the environment : assessment of doses to man",*
- *ICRP Publication 39 "Principles for limiting exposure of the public to natural sources of radiation",*

- *ICRP Publication 63 "Principles of intervention for protection of the public in a radiological emergency",*
- *ICRP Publication 82 "Protection of the public in situations of prolonged radiation exposure",*

or for environmental exposure:

- *ICRP Publication 91 "A Framework for Assessing the Impact of Ionizing Radiation on Non-Human Species"*

Another commission whose work directly influences radiation protection is the International Commission on Radiation Units (ICRU). This commission defines the parameters and units used in the field of radiation protection.

Still at the international level, the International Atomic Energy Agency (IAEA) publishes safety norms applicable to operations placed under agency supervision. The Basic Safety Standards (BSS)in protection against ionizing radiation and safety of radioactive sources were published in 1996 by the IAEA. These define the requisite conditions for protecting human beings in all areas involving exposure to radiation.

We can also cite the Scientific Committee at the United Nations for their study on the effects of ionizing radiation (UNSCEAR) and whose voluminous reports constitute the basis of our knowledge in this field.

Other bodies conduct studies and publish reports in the field of radiation protection and their conclusions influence legislators in their work to prepare regulations associated with radiation protection. Some examples include the Radiation Protection Division of the Health Protection Agency (HPA) in England, the National Council on Radiation Protection and Measurements (NCRP) in the United States, the Centre d'Etudes et de Protection Nucléaire (CEPN) in France, and the Bundesamt für Strahlenschutz (BfS) in Germany.

Numerous scientific projects are also conducted within the framework of professional organizations associated with the International Radiation Protection Association (IRPA). At this level, Switzerland is represented by the Fachverband für Strahlenschutz (FS), which includes the Association Romande de Radioprotection (ARRAD).

2. Swiss situation

The Law of 22 March 1991 on Radiation Protection (LRaP), the Ordinance of 22 June 1994 on Radiation Protection (ORaP) both became legal on 1 October 1994. Added to these legal documents are a certain number of Ordinances or Departmental directives. Remember that these regulations are based on the latest international recommendations, and in particular, on Publication 60 from the ICRP from 1990.

Although most of the themes concerning radiation protection are dealt with in the law on radiation protection, they are quite general in nature and so responsibility is given to the Swiss Federal Council for enacting application measures.

Specific aspects of the diverse applications have not been integrated into ORaP. These are subject to departmental ordinances.

Legislative functioning established in the field of radiation protection (see Table 6.1.) is also well structured.

The texts of these documents can be obtained on the web sites of the FOPH (www.bag.admin.ch), of METAS (www.metas.ch) or the Federal Council (www.admin.ch).

Table 6.1: Legislative functioning in radiation protection (the list of documents is not exhaustive)

Level	Legislative text in radiation protection	Additional legislation
Law	Law of 22 March 1991 on radiation protection (LRaP)	Law of 23 December 1959 on atomic energy and federal order of 6 Oct 1978 Law of 18 March 1983 on responsibility in nuclear matters Federal ordinance of 17 April 1985 on transporting dangerous goods by road.
Federal ordinance	Ordinance of 22 June 1994 on Radiation protection (ORaP)	Ordinance of 26 June 1991 on organization and intervention in case of a radioactivity increase (OROIR)
Departmental Ordinances	Ordinance of 15 Sept 1998 on training and authorized activities in radiation protection Ordinance of 7 Oct 1999 on personal dosimetry Ordinance of 20 January 1998 on radiological installations used in medicine Ordinance of 15 Nov 2001 on using sealed radioactive sources in medicine Ordinance of 15 Dec 2004 on radiation protection as applied to electron accelerators used in medicine Ordinance of 31 Jan 2001 on non-medical installations which produce ionizing radiation Ordinance of 21 Nov 1997 on using open radioactive sources Ordinance of 1 July 1992 on distributing iodine tablets to the population Ordinance of 8 July 1996 on radioactive waste necessitating delivery Ordinance of 24 March 1999 on emoluments received in the field of radiation protection	Directive from the Swiss Federal Office of Public Health (FOPH) in the field of ionizing radiation Directives from the Federal Metrology Department (METAS) on verifying radiation protection instruments

Field	Notices / FOPH Directives in the field of radiation protection
Laboratories	Radiation protection in the schools (L-02-01) Radioactive minerals (L-02-2) Therapeutic applications of I-131 (L-04-01) Marking and biopsy of sentinel lymph node ganglions (L-04-02) Dosimetry when using open radioactive sources (L-06-01) Calculation of the shielding required for a PET facility (L-07-01) Calculation of the shielding required for patient rooms in I-131 therapy (L-07-03) Guideline values for ambient dose rates in nuclear medicine departments (L-07-04) Monitoring the stability of activimeters (L-09-01) Information notice concerning 65 mg Potassium Iodide
X Rays	Ordinance on emoluments (R-02-01) Measurements relating to replacing radiological installations (R-04-01) Declaration form for companies plus verbal process for the periodic monitoring in radiation protection (R-04-02) Quality assurance for suppliers of radiological devices (R-06-01) Dosimetric monitoring in hospitals (R-06-03) Signage for radiology rooms (R-07-01) Total filtration for radiological installations used in medical diagnostics (R-08-01) Monitoring the quality of mammography installations (R-08-02) Diagnostic reference levels for radiological exams (R-08-04) Quality control for digital dental radiography (R-08-05) Quality control for digital radiography systems (R-08-06) Sensitivity classification for screen-film combinations and medical imaging systems ((R-09-01) Protection for patients, personnel and third parties for diagnostic exams(R-09-02) Manual film development (R-09-03)

METAS Directives in the field of radiation protection
Directive concerning the construction, metrological characteristics, verification and testing of radiation protection instruments used for measuring external radiation.
Directives concerning the construction, metrological characteristics, verification and testing of surface contamination monitors.
Directives concerning the construction, metrological characteristics and verification of dosimetric systems of mobile references used in radiotherapy.
Directives concerning the construction, metrological characteristics, verification and testing of dosimeters used in radiodiagnostics and measuring instruments used for testing radiodiagnostic installations.
Directives concerning the construction, metrological characteristics and testing, through verification or intercomparison, of devices for measuring activity (dose calibrators) used in nuclear medicine laboratories.

3. Law and Ordinance on Radiation Protection

The law is structured in the following manner

Chapter 1: General Provisions
Chapter 2: Protecting humans and the environment
Chapter 3: Authorizations and monitoring
Chapter 4: Civil liability and insurance
Chapter 5: Procedure, claims and emoluments
Chapter 6: Penal provisions
Chapter 7: Final provisions

For its part, the Ordinance has the following structure:
Chapter 1: General provisions and principles of radiation protection
Chapter 2: Technical qualification, officers, training and perfecting knowledge
Chapter 3: Medical applications of radiation
Chapter 4: Protecting individuals exposed to radiation
Chapter 5: Using installations and radioactive sources
Chapter 6: Radioactive waste
Chapter 7: Failures
Chapter 8: Environmental monitoring and food supplies
Chapter 9: Protecting the population in case of radioactivity increase
Chapter 10: Authorizations and monitoring
Chapter 11: Penal and final provisions

4. Summary:

- *Radiation protection is based on international recommendations or regulations coming in particular from:*
  - *the International Commission on Radiological Protection (ICRP);*
  - *The International Commission on Radiation Units (ICRU)*
  - *The International Atomic Energy Agency (IAEA)*
- *Radiation Protection is governed in Switzerland by:*
  - *Law of 22 March 1991 on radiation protection*
  - *Ordinance of 22 June 1994 on Radiation protection*
  - *various technical ordinances:*
    - *Ordinance on training in radiation protection*
    - *Ordinance on personal dosimetry*
    - *ordinance on using open radioactive sources*
    - *Ordinance on non-medical X-ray installations*
    - *Ordinance on radioactive waste*
    - *Directives coming from overseeing federal offices: FOPH and METAS.*
- *The main points of the regulations are:*
  - *fixing exemption limits*
  - *technical qualification and training of individuals with occupational exposure to ionizing radiation*
  - *dosimetry and dose limits for individuals with occupational exposure to ionizing radiation*
  - *conditions of use for radiation sources, including quality assurance*
  - *management of radioactive waste*
  - *measures to take in case of failure*
  - *environmental monitoring, including radon and food supplies*
  - *protecting the population in case of radioactivity increase*
  - *authorization process and monitoring of departments*

5. Practice questions

1. Which legal text defines the principle of dose limits?

2. In which document can you find requirements for outfitting a type C working area?

3. How does the ordinance on radiation protection supervise the dosimetric monitoring of individuals with occupational exposure to ionizing radiation?

4. How do you know whether a radioactive source is subject to the law on radiation protection?

5. Which documents must the radiation protection officer possess in a nuclear medicine department with a type B laboratory?

6. Which federal department deals with the technical ordinances in the field of radiation protection?