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NSEDA - WEAPON

Module 2 - Radiation Hazards and Protection

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Student Name:



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Abbreviations and Acronyms

ADS	Approved Dosimetry Service
ALARA	As Low As Reasonably Achievable
ALARP	As Low As Reasonably Practicable
ALI	Annual Limit of Intake
ARS	Acute Radiation Syndrome
AWE	Atomic Weapons Establishment
BAT	Best Available Techniques
BPM	Best Practicable Means
DDREF	Dose and Dose Rate Effectiveness Factor
DNA	Deoxyribonucleic Acid
DRPS	Defence Radiation Protection Service
EA	Environment Agency
EASR18	Environmental Authorisations (Scotland) Regulations 2018
EPD	Electronic Personal Dosemeter
EPR16	Environmental Permitting Regulations 2016
GPM	Gaussian Plume Model
HSE	Health and Safety Executive
HVL	Half Value Layer
ICRP	International Commission on Radiological Protection
IRR17	Ionising Radiations Regulations 2017
LD	Lethal Dose
NIEPA	Northern Ireland Environmental Protection Agency
ONR	Office for Nuclear Regulation
PADC	Polymerisation of Allyl Diglycol
PED	Personal Electronic Dosemeter
PPE	Personal Protective Equipment
RBE	Relative Biological Effectiveness
REPP19	Radiation (Emergency Preparedness and Public Information) Regulations 2019
RPA	Radiation Protection Adviser
RPE	Respiratory Protective Equipment
RPS	Radiation Protection Supervisor
RSA93	Radioactive Substances Act 1993
SEPA	Scottish Environment Protection Agency
TIAC	Time Integrated Air Concentration
TLD	Thermoluminescent Dosemeter

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Chapter 1

Types and Properties of Radiation

1.1 Aims

By the end of this chapter the student should be able to

- explain the difference between ionising and non-ionising radiation and also between electromagnetic and particulate radiations
- describe the origin, physical properties and penetrating characteristics of alpha, beta, gamma and neutron radiation
- define 'activity' and state its SI unit
- define 'absorbed dose', 'equivalent dose' and 'effective dose' and state their units, explain the need for these quantities and the relationship between them with reference to the radiation weighting factors and tissue weighting factors
- define the quantity 'collective dose' and state its unit.

1.2 Ionising and Non-Ionising Radiation

Ionisation is the removal (or addition) of an orbital electron from an atom, as shown in Figure 1.1, to create an ion. Positive ions are produced when an electron is removed, and negative ions are produced following the addition of an electron.

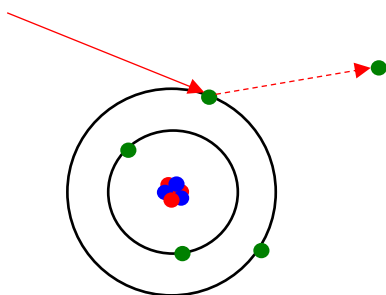


Figure 1.1 - Ionisation resulting in the production of a positive ion

Ionisation energy is the amount of energy required to cause ionisation. Typical ionisation energies range from a few eV to tens of eV¹. Radiation with sufficient energy to cause ionisation is termed **ionising radiation**.

¹The electronvolt (eV) is the amount of energy gained (or lost) by a single electron moving across an electric potential difference of 1 volt. 1 eV is therefore equal to 1.6×10^{-19} J.

1.3 Electromagnetic Radiation

Electromagnetic radiation is the movement of energy, through space or a medium, and is composed of both electric and magnetic waves.

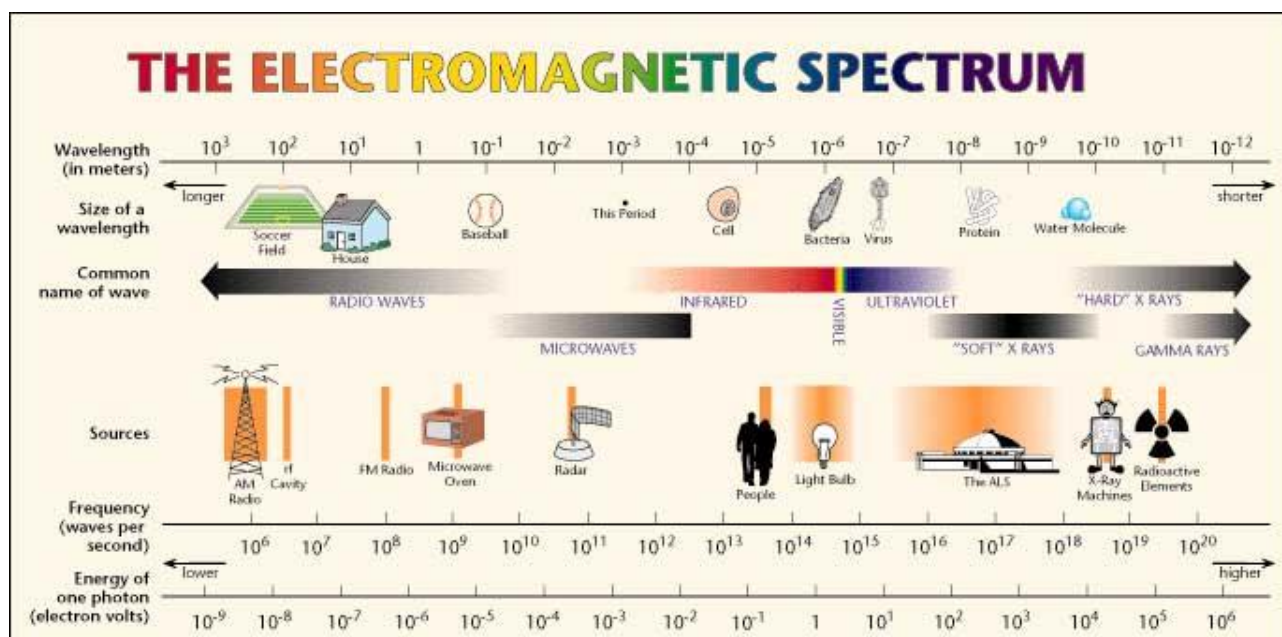


Figure 1.2 - The electromagnetic spectrum

There are various types of electromagnetic radiation, with varying wavelengths, ranging from thousands of metres to fractions of a metre (gamma rays). All the different types of electromagnetic waves travel at the same speed (the speed of light $\sim 3.0 \times 10^8 \text{ ms}^{-1}$) through space.

The frequency and wavelength of electromagnetic waves are related by

$$\text{Wavelength (m)} \times \text{Frequency (s}^{-1}\text{)} = \text{Speed (ms}^{-1}\text{)} \quad 1.1$$

The energy of an electromagnetic wave is given by

$$\text{Energy (J)} = \text{Plank's constant (Js)} \times \text{Frequency (s}^{-1}\text{)} \quad 1.2$$

Example 1.1

What is the frequency and energy (in J and eV) of an x-ray with a wavelength of 10 nm?

Solution

$$\text{Frequency} = 3 \times 10^8 / 10^{-8} = 3 \times 10^{16} \text{ Hz}$$

$$\text{Energy} = 6.63 \times 10^{-34} \times 3 \times 10^{16} = 1.99 \times 10^{-17} \text{ J} = 1.99 \times 10^{-17} / 1.6 \times 10^{-19} \text{ eV} = \mathbf{124 \text{ eV}}$$

The only types of electromagnetic radiation with sufficient energy to cause ionisation ($\sim$ few eV), and hence described as ionising radiation, are x-rays and gamma rays. All other types of electromagnetic radiation, e.g. radio waves and visible light, are described as non-ionising radiation.

1.4 Particulate Radiations

Particulate radiation is the radiation of energy by means of fast moving subatomic particles. The particles can be charged (alpha particles, beta particles) or uncharged (neutrons). Particulate radiations generally have sufficient energy to cause ionisation.

1.5 Types of Ionising Radiation

The main types of ionising radiation are:

Alpha (α) radiation consists of two protons and two neutrons (i.e. a helium nucleus). Thus, alpha particles are charged particles and massive by nuclear standards.

Beta (β) radiation consists of high-speed electrons ejected from the nucleus of a radioactive atom. It is also a charged particle but much lighter than the alpha particle.

Gamma (γ) radiation consists of high energy electromagnetic waves emitted by excited nuclei.

X-radiation (X-rays) consists of high energy electromagnetic waves emitted when atomic electrons undergo a change in shell.

Neutron radiation consists of neutrons emitted from a radioactive nucleus or produced during certain nuclear reactions. Neutrons are emitted with widely varying energies from fractions of an eV to greater than an MeV. For convenience, they are divided into two main groups: fast and thermal.

Table 1.1 gives the types and penetrating properties of various types of radiation.

Table 1.1 - Summary of Types and Properties of Radiation

Radiation		Origin	Mass (amu)	Charge/e	Range	
					Air	Tissue
Alpha		Radioactive decay	4	+ 2	cm	μm
Beta		Radioactive decay	1/1820	± 1	metres	mm
Gamma		Radioactive decay	0	0	up to 100s m	very penetrating
X-Ray		Atomic Electrons	0	0	up to 100s m	very penetrating
Neutrons	Fast	Fission	1	0	up to 100s m	very penetrating
	Thermal	Moderation	1	0	less than fast neutrons	very penetrating

1.6 Activity

The activity of a substance is defined as the number of nuclei disintegrating per second. Activity is measured in becquerels.

Unit becquerel, Bq

Quantity measures rate of disintegration of nuclei

Definition one becquerel equals one disintegration per second.

1.6.1 Specific Activity

The specific activity is a measure of the activity per unit mass as given in Equation 1.3. It is not only used for well characterised materials but also for radioactive waste streams.

$$\text{Specific activity} = \frac{\text{Activity}}{\text{Mass}} \quad 1.3$$

Example 1.2

The specific activity of ^{226}Ra is $3.7 \times 10^{10} \text{ Bqg}^{-1}$. What is the activity of 1 g of ^{226}Ra ?

Solution

Activity = Specific activity x mass = $3.7 \times 10^{10} \times 1 = 3.7 \times 10^{10} \text{ Bq}$

Example 1.3

The specific activity of ^{235}U is $1.23 \times 10^4 \text{ Bqg}^{-1}$. What is the mass of 10^{10} Bq of ^{235}U ?

Solution

Mass = Activity / Specific activity = $10^{10} / 1.23 \times 10^4 = 8.13 \times 10^5 \text{ g} = 813 \text{ kg}$

Unit	becquerel per gram, Bqg^{-1}
Quantity	measures activity per unit mass
Definition	$1 \text{ Bqg}^{-1} = 1 \text{ disintegration per second per gram}$

1.7 Absorbed Dose

The concept of absorbed dose (D) is applicable for all types of radiation (particulate and electromagnetic) and all target materials. It is a measure of energy deposition in any medium by any type of ionising radiation. The absorbed dose at a point is defined as

$$D = \frac{d\bar{\epsilon}}{dm} \quad 1.4$$

where $d\bar{\epsilon}$ is the mean energy deposited by the radiation in matter of mass dm .

The SI unit of dose is the gray (Gy) and is defined as an energy deposition of 1 Jkg^{-1} .

The absorbed dose is the fundamental dosimetric quantity in radiological protection and is defined in terms that allow it to be specified at a point. However, unless otherwise stated, when referring to absorbed doses to organs or tissues, it refers to the absorbed dose over the whole organ. It should be noted that absorbed dose is the energy deposited by the radiation. This is significant, especially when considering radiations that give rise to bremsstrahlung², since the point of energy deposition may be remote from the point of energy transfer.

² Electromagnetic radiation produced by the deceleration of a charged particle when deflected by another charged particle, e.g. an electron by a nucleus – see Section 4.2.3.2

The absorbed dose is heavily dependent on the target material. For example, if the radiation is not completely stopped within the volume of the material, then only a fraction of incident radiation energy will be transferred to the target material. This fraction depends upon the nature of the incident radiation (alpha, beta etc.), the energy of the incident radiation (the greater the energy the further it will travel), and the atomic composition of the target material (elemental composition and atom density).

Consider a beam of X-rays incident upon a person's chest. The absorbed dose will be different for the soft tissue, the rib cage and the air within the lungs. If this was not the case, and the X-rays were stopped uniformly, imaging using X-rays would not be possible.



Figure 1.3 - X-rays of broken bones

1.8 Radiation Weighting Factors and Equivalent Dose

The absorbed dose is a useful physical concept, but it should be noted that in biological systems the same degree of damage is not necessarily produced by the same absorbed dose from different types of ionising radiation.

This difference in relative biological effectiveness (RBE) must be accounted for if we wish to combine doses of different radiations to obtain the total biologically equivalent dose. Each radiation type is assigned a factor that reflects its RBE of causing **stochastic effects at low doses** (see Section 2.7). This factor is called the radiation weighting factor, w_R .

Values of w_R have been determined by the International Commission on Radiological Protection³ (ICRP) and published in ICRP 103 'The 2007 Recommendations of the International Commission on Radiological Protection'. The radiation weighting factors currently in use are given in Table 1.2.

The equivalent dose in a tissue T H_T , is given by Equation 1.5:

$$H_T = \sum_R w_R D_{T,R} \quad 1.5$$

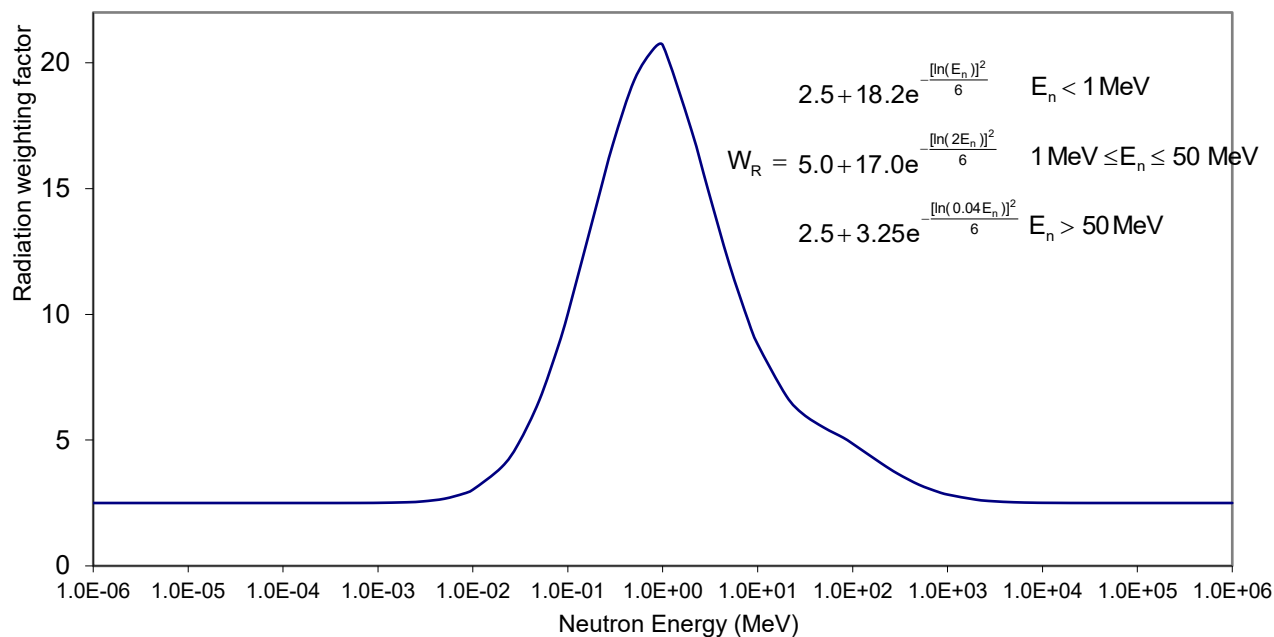
where $D_{T,R}$ is the absorbed dose averaged over the tissue/organ T, due to radiation R.

The unit of equivalent dose is the sievert (Sv).

³ The International Commission on Radiological Protection (ICRP) is an independent, international organisation with the mission to provide recommendations and guidance on radiation.

Table 1.2 - Radiation Weighting Factors

Radiation Type	ICRP 103 Radiation Weighting Factor, w_R
Photons	1
Electrons and muons	1
Protons and charged pions	2
Alpha particles, fission fragments, heavy ions	20
Neutrons	A continuous curve as a function of energy (see Figure 1.4)

Figure 1.4 - Radiation weighting factor for neutrons as a function of neutron energy E_n , in MeV

When a radiation field is composed of different radiation types and/or energies with different values of w_R , the absorbed dose should be split into blocks each with its own w_R value and summed to give the total equivalent dose.

Example 1.4

In one year, a person receives a whole body absorbed dose of 2 mGy of gamma radiation, 0.5 mGy of alpha radiation and 0.1 mGy of thermal (energy = 0.025 eV) neutrons. What is their equivalent dose?

Solution

Radiation	Absorbed Dose	Radiation Weighting Factor	Equivalent Dose
Gamma	2 mGy	1	2 mSv
Alpha	0.5 mGy	20	10 mSv
Thermal neutrons	0.1 mGy	2.5	0.25 mSv
Total			12.25 mSv

1.9 Tissue Weighting Factors and Effective Dose

It is well known that radiation is much more likely to cause some cancers, such as leukaemia or bone cancers, than other cancers, such as skin cancer. By adjusting each individual organ dose by a factor to reflect the radiosensitivity of that particular organ, it is possible to estimate the total harm caused by the exposure.

The factor by which the equivalent dose to an organ or tissue is weighted is called the **Tissue Weighting Factor (w_T)**. This represents the relative contribution of that organ or tissue to the total detriment and as such, the sum of all the tissue weighting factors is one. This tissue weighted equivalent dose (in effect a doubly weighted absorbed dose) is called the **Effective Dose**.

In considering the relative contributions of various organs and recognising that the process of deriving them and the original data both contained large uncertainties, the ICRP decided that the values for the relative contribution could be rounded. These assigned Tissue Weighting Factors are available from ICRP 103 and reproduced in Table 1.3.

The effective dose, E , is a measure of the whole-body detriment caused by exposure to ionising radiation and can be calculated using Equation 1.6.

$$E = \sum_T w_T H_T \quad 1.6$$

where H_T is the equivalent dose in tissue or organ T and w_T is the weighting factor for tissue T .

Both effective dose and equivalent dose have the unit of the sievert. Most of the time radiation protection is concerned with the whole-body detriment and hence the effective dose. This is the quantity assessed by personal dosimeters and monitoring instruments. However, sometimes it is important to know the equivalent dose in which case the exposed part of the body should be clearly stated. For example, following inhalation of ^{131}I the crucial dose is the equivalent dose to the thyroid.

When a non-uniform irradiation occurs (and almost all exposures are non-uniform in practice), the equivalent doses to each tissue and organ should be determined. The effective dose is calculated by summing all the doses obtained by multiplying each equivalent dose by the appropriate w_T value. This reflects the harm to the whole body and allows the effects of different non-uniform exposures to be compared.

Example 1.5

Following an intake of plutonium a person receives an equivalent dose of 100 mSv to both the lung and the bone marrow, what is the effective dose?

Solution

Effective Dose = $w_{\text{lung}} \times H_{\text{lung}} + w_{\text{bone marrow}} \times H_{\text{bone marrow}}$

$0.12 \times 100 + 0.12 \times 100 = \mathbf{24 \text{ mSv}}$

Table 1.3 - Tissue Weighting Factors

Organ /Tissue	Tissue Weighting Factor, w_T
Bone Surface	0.01
Skin	0.01
Bladder	0.04
Breast	0.12
Liver	0.04
Oesophagus	0.04
Thyroid	0.04
Bone Marrow	0.12
Colon	0.12
Lung	0.12
Stomach	0.12
Gonads	0.08
Brain	0.01
Salivary Glands	0.01
Remainder	0.12
TOTAL	1.0

Example 1.6

A person is exposed to a narrow beam of thermal neutrons from a research reactor. The doses to the person are as shown below. Calculate the effective dose to the person.

skin - 40 mGy, oesophagus - 12 mGy, thyroid - 9 mGy

Solution

Use $w_R = 5$ for thermal neutrons.

Tissue	Dose (mGy)	H (mSv)	w_T	Effective dose $w_T \times H$ (mSv)
Skin	40	$40 \times 2.5 = 100$	0.01	1
Oesophagus	12	$12 \times 2.5 = 30$	0.05	1.5
Thyroid	9	$9 \times 2.5 = 22.5$	0.05	1.125
			$E_T =$	$1+1.5+1.125 = \mathbf{3.625 \text{ mSv}}$

1.10 Collective Effective Dose

The collective effective dose is defined as the integral over effective doses received by the population over a specified time period as shown in Equation 1.7.

$$S(E_1 E_2 \Delta T) = \int_{E_1}^{E_2} E \left[\frac{dN}{dE} \right]_{\Delta T} dE \quad 1.7$$

where $\left[\frac{dN}{dE} \right]_{\Delta T} dE$ is the number of individuals who are exposed to an effective dose between E and $E + dE$ within time period ΔT .

This can be approximated to

$$S = \sum_i E_i N_i \quad 1.8$$

where E_i is the average effective dose for a subgroup i , and N_i is the number of individuals in this subgroup.

Collective effective dose is generally shortened to collective dose, as it is recognised that collective equivalent dose is only likely to be used in special circumstances. In the case of occupational exposure collective dose can be used as part of the optimisation process, although it does have some limitations especially when the dose distribution is not uniform. It can also be used as a 'performance indicator' to characterise the total exposure associated with the operation of installations over a given time period.

Similarly, collective dose can be used as part of the optimisation process in the case of public exposure e.g. from radioactive discharges, but again it does have its limitations and must be used with caution. ICRP 103 explicitly states that collective dose is not intended as a tool for epidemiological risk assessment, and it is inappropriate to use it in risk projections. The aggregation of very low individual doses over extended time periods is inappropriate and, in particular, the calculation of the number of cancer deaths based on collective doses from trivial individual doses should be avoided.

1.11 Summary

Ionisation - removal of an orbital electron from an atom

Ionising radiation - electromagnetic or particulate radiation with sufficient energy to cause ionisation

Absorbed dose - energy deposition in any medium by any type of ionising radiation ($1 \text{ Gy} = 1 \text{ Jkg}^{-1}$)

Radiation weighting factor w_R - a measure of the ability of a particular type of radiation to cause biological damage, which is related to the density of ionisation

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

Tissue weighting factor w_T - a factor reflecting the radiosensitivity of a particular tissue or organ

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

Collective dose (shortened from collective effective dose) - the total dose for a group of individuals

1.12 Review Questions

1. The specific activity of 100 litres of water containing 1 kBq of activity is:
 - a. 100 μGy
 - b. 10 Bq
 - c. 10 Bqkg^{-1}
 - d. 0.01 Bqkg^{-1}
2. Which of the following **is not** measured in sieverts?
 - a. Effective dose
 - b. Equivalent dose
 - c. Committed effective dose
 - d. Collective dose
3. A person's thyroid gland is exposed to a field of neutrons. The fast (1 MeV) neutron dose is 1 mGy and the thermal (<10 keV) neutron dose is 10 mGy. The equivalent dose to the thyroid is about:
 - a. 45 mSv
 - b. 11 mGy
 - c. 45 mGy
 - d. 11 mSv
4. Following an intake of plutonium, a person receives the following committed equivalent doses:
100 μSv to the lung ($w_T=0.12$)
50 μSv to the red bone marrow ($w_T=0.12$)
20 μSv to the bone surface ($w_T=0.01$)
The committed effective dose to the person is:
 - a. 364 μSv
 - b. 18.2 μSv
 - c. 162 μSv
 - d. 3.4 mSv
5. A source has an activity of 5750 kBq. What is the activity in MBq?
 - a. 575 MBq
 - b. 57.5 MBq
 - c. 5.75 MBq
 - d. 0.575 MBq
6. What is a physical quantity? Which quantities from the list below are physical quantities. Explain your answer
Specific activity
Activity
Absorbed dose
Effective dose
Equivalent dose

7. Assuming ionising radiation has an energy of at least 1.6×10^{-18} J list all of the types of electromagnetic radiations that can be classed as **non-ionising**.
8. What is the difference between gamma radiation and X-rays?
9. A work package to clean a radioactive component involves the following whole body exposures:

Worker A: 1 mGy thermal neutrons and 2 mGy gamma

Worker B: 0.5 mGy thermal neutrons and 1 mGy gamma

Worker C: 0.1 mGy thermal neutrons and 2 mGy gamma

What is the collective dose for the work package?

- a. 6.6 man mGy
- b. 6.6 man mSv
- c. 9 man mSv
- d. 9 man mGy

Chapter 2

Biological Effects of Radiation

2.1 Aims

By the end of this chapter the student should be able to

- describe in general terms the effects of ionising radiation on biological systems
- describe the key characteristics of tissue reactions, identify the cells affected and the symptoms arising
- define the terms 'acute' and 'chronic', 'early' and 'late' and 'somatic' and 'hereditary'
- describe how cell damage can lead to cancer and outline the basis for cancer risk estimates
- describe why sex cell damage can lead to hereditary effects
- describe the key characteristics assumed for stochastic effects and state approximate values for the risk per Sv of cancer and hereditary effects.

2.2 Introduction

The interaction of ionising radiation with the human body, arising either from sources outside the body or from internal contamination of the body by radioactive substances, leads to biological effects which may later show up as clinical symptoms. The principal effect of the radiation is to cause ionisation and the biological effects are a consequence of this ionisation. The nature and severity of these symptoms and the rate at which they appear depends upon the **amount** of radiation absorbed and the **rate** at which it is received.

2.3 Cells

All living creatures and organisms consist of tiny structures known as cells. A typical cell is 10 μm in diameter, although some cells are only 3 μm in diameter and others, such as nerve cells, have diameters of 100 μm and branches which extend up to 1 m in length. Although cells vary a great deal in shape, size and specific function they all possess some general features.

A simplified cell consists of three distinct regions as shown in Figure 2.1. It consists of a **nucleus** and a surrounding liquid known as the **cytoplasm**. The cytoplasm is contained within the cell **membrane** and its main function is in the building up of protein and other essential materials and in the breaking down of other material to provide energy. The nucleus directs cellular activities including the metabolic functions of the cytoplasm. The apparatus for directing the cytoplasm

consists of **chromosomes**. There are 46 chromosomes in human cells except for the ova and sperm in which there are 23. These chromosomes only become distinguishable during cell division.

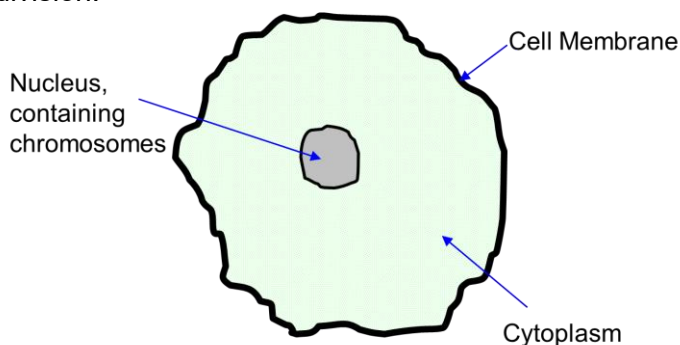


Figure 2.1 - Simplified cell structure

All chromosomes carry **genes** which in turn are made up of highly complex **DNA molecules**. Every cell in the body has the same set of genes. Each gene is a set of instructions for building up a complex molecule, such as a protein or enzyme. Sets of genes vary from individual to individual.

2.3.1 Cell Reproduction

Cells reproduce to enable the organism to grow and to compensate for cells which die. The lifespan and rate of reproduction of different types of cells vary from a few hours to many years. The rate of cell reproduction is also variable for a given cell type: cell reproduction may be stimulated by hormonal or other factors. Reproduction occurs in two ways - **mitosis and meiosis**.

Mitosis

Mitotic cells are ordinary cells in the body and a single one of these cells splits to produce two identical daughter cells. The various stages of the cell reproduction are shown in Figure 2.2.

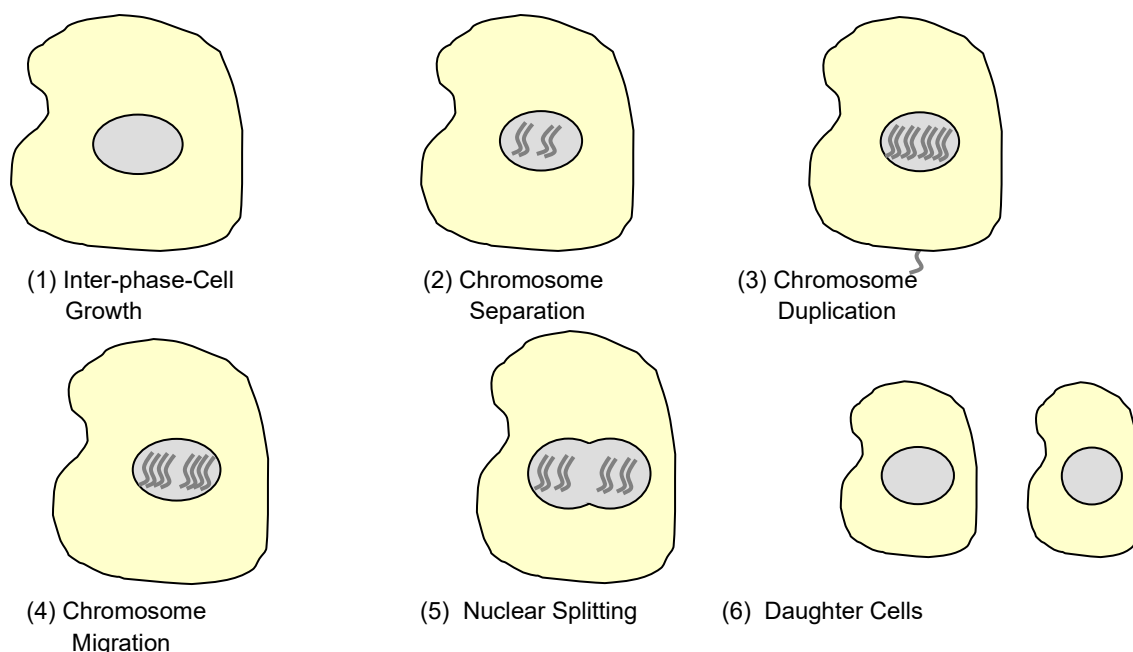


Figure 2.2 - Cell reproduction cycle (mitosis)

Meiosis

Meiosis is a special kind of reproduction which occurs in the male sperm cells and female ovum cells. In these cells, the number of chromosomes is half of the mitotic cells, such that when a sperm cell and an ovum cell unite there will be a full complement of chromosomes, half contributed by each parent.

2.4 Interaction of Radiation with Cells

Radiation incident upon a cell creates the same events on the atomic scale as would occur within any other material, namely excitation and ionisation. However, from a biological aspect the crucial feature is how the net consequences of these processes are expressed at the cellular level.

The radiation can damage the cell **directly or indirectly**, depending which molecules within the cell become ionised. Direct damage is ionisation of key molecules, such as the DNA, within the cell. Indirect damage is ionisation of molecules within the cell to generate highly reactive chemicals that harm the cell. Indirect damage is much more probable than direct damage since the main component of the cell is water. The ionisation of water may result in the production of hydrogen peroxide (H_2O_2) and free hydroxide radicals ($\text{OH}^\bullet$).

There are three important consequences following cellular damage caused by radiation (acting directly or indirectly) – repair, death or mutation, as shown in Figure 2.3.

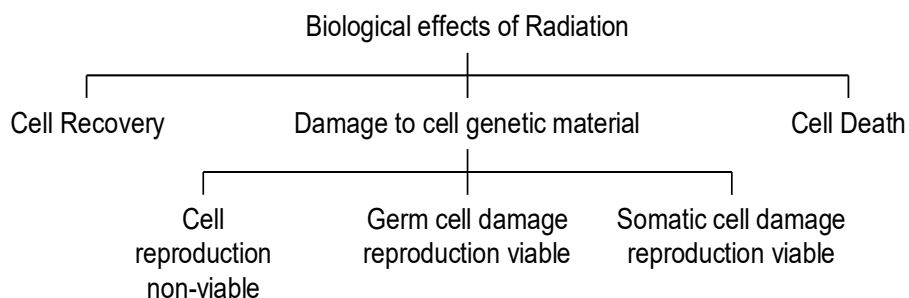


Figure 2.3 - Cellular effects of ionising radiation

2.4.1 Cell Recovery - DNA Repair

To deal with DNA damage, cells have evolved complex, enzyme-mediated repair systems. These systems are specific for different molecular forms of DNA damage, resulting in the induced DNA lesions being identified and removed, often within a timescale of tens of minutes. These repair mechanisms generally rely on only one strand of the DNA double helix being broken so that the cell 'knows' what repair is required.

2.4.2 Cell Death

During **interphase** cells are very **radio-resistant** and quite high doses are required to kill them (≈ 10 Gy). This means that cells that have lost the power of division, for example nerve, muscle and some bone cells, are very radio resistant. An exception to this is the lymphocytes, which are more easily killed during interphase.

2.4.3 Damage to Cell Genetic Material

Cell Reproduction Non-Viable

Where the DNA damage leads to an inability of the cell to reproduce, the cell population of those organs or tissues in which the cells are most rapidly dividing are going to be most affected, i.e. those which have the shortest life or are being most rapidly used. Typical cell life spans are listed in Table 2.1. It is the effects of high doses of radiation on those cells with short life spans that produces the characteristic symptoms of radiation sickness.

Germ Cell / Somatic Cell Damage Reproduction Viable

Occasionally the damage to the chromosome is such that it will not prevent successful cell division. It may be that only one or two genes have been damaged. Generally, this will only affect the daughter cells and not have any great effect on the body. If there are several double strand breaks to the DNA it is likely that the gene will not be repaired or will be mis repaired. It is then probable that such small mutations are the cause of some cells eventually being unable to control division and thus becoming cancerous. Where the affected cell is a germ cell (sperm or ovum), then this could result in hereditary disorders in succeeding generations and give rise to heritable effects (see Section 2.7.3).

Table 2.1 - Cell Lifespan

Cell Type	Lifespan
White blood cell	2 - 3 days
Gastro-intestinal tract lining	~ 15 days
Red blood cells	~100 days
Bone cells	~ 7 years
Brain cells	Lifetime of individual

2.5 Definitions

When the overall effects of radiation on the whole body is considered rather than effects on individual cells or tissues, effects of exposure may be categorised as:

- **Acute** - the exposure is large and received over a short period of time
- **Chronic** - the exposure is at a low level over a long period of time.

Radiation effects can be divided into two classes:

- **Somatic** - damage appears in the irradiated person
- **Hereditary** - damage appears in future generations.

Somatic effects can also be divided into two categories:

- **Early Effects** - these effects appear shortly after exposure and are clinically identifiable immediately
- **Late Effects** - these effects may not become apparent for many years.

2.6 Tissue Reactions

When the absorbed dose to an organ or tissue is high, the radiation damage may become apparent in a short period of time. This somatic, early effect is the main type of **tissue reaction**. For these effects to occur, the absorbed dose must exceed a certain minimum value, known as the **threshold**. Once the threshold for a particular effect has been reached, the severity of the effect increases with increased dose. A graphical representation of tissue reactions is given in Figure 2.4.

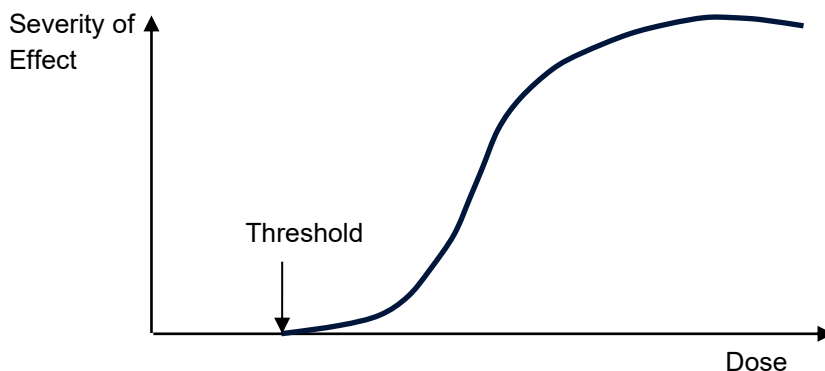


Figure 2.4 - Tissue reactions

Examples of early tissue reactions are acute radiation effects from whole body doses and significant skin doses (see Section 2.6.2). An example of a tissue reaction which may not occur for many years is damage to the lens of the eye, resulting in observable opacities in the lens or, in extreme cases, visual impairment as the result of a cataract.

A large whole body radiation exposure will produce early symptoms (see Section 2.6.1) in an individual that are deterministic in nature and present as **tissue reactions**. These early effects of an acute exposure are sometimes referred to as radiation sickness or acute radiation syndrome.

For large doses the decrease in cell numbers will lead to clinically observable pathological conditions such as loss of tissue function or a consequential reaction as the body attempts to repair the damage. Where the tissue is vital enough and damaged sufficiently, the end result could be death (for example, following the destruction of bone marrow after whole body irradiation). Simultaneous damage to several tissues could exacerbate the situation. For healthy individuals the threshold dose (defined as the dose to produce a defined, specific effect in at least 1% of individuals exposed) for causing detectable clinical harm will be up to a few Gy, depending upon the tissue irradiated. Above this threshold, the severity of the harm increases with dose, as discussed previously, reflecting the increased percentage of cells damaged within the tissue or organ.

If some individuals in the exposed group are already in a state of health approaching a relevant pathological condition, they will reach that condition as a result of exposure to radiation after a smaller loss of cells than would otherwise be the case. It should also be noted that there is also a variation in sensitivity amongst healthy individuals in a population. The organs/tissues/biological systems most affected following an acute exposure are the bone marrow, skin, gastro-intestinal tract and the central nervous system. Threshold doses for tissue reactions effects for each of these is given in Table 2.2.

Table 2.2 - Threshold Doses

System	Threshold Whole Body Dose (Gy)
Bone marrow (blood)	0.5
Skin	3
Gastrointestinal tract	5
Central nervous system	10

An estimate can be made of the dose which would be lethal for 50% of the exposed individuals. This is referred to as the LD₅₀. This is further expanded to indicate a time period for death to occur e.g. the LD_{50/60} is the dose which could be lethal for 50% of the exposed population in 60 days and is generally considered to have a value of 3-5 Gy for man.

2.6.1 Symptoms of Acute Whole Body Exposure

The first symptoms of an acute radiation exposure are nausea and vomiting. The onset is normally within 24 hours, although the larger the exposure, the sooner it will occur. For absorbed doses of less than 1 Gy, this symptom may not appear and for absorbed doses of less than 0.5 Gy it is unlikely.

This first symptom is followed by a latent period, lasting from about two to ten days depending on the magnitude of the absorbed dose, when there appears to be no ill effects. This is because the lack of replacement cells for the short-lived body cells (white blood cells, cells lining gut etc.) has not yet become apparent. During this period the individual feels well. After the latent period, the main effects of radiation sickness appear, with causes as listed in Table 2.3 and given in more detail in Table 2.4.

Table 2.3 - Radiation Sickness Symptoms and Causes

Name and Timescale	Symptoms	Cause
Prodromal (initial) reaction: within a few hours	Nausea and vomiting	Nervous reaction to loss of cells
Latent period: a few days	None	
Main stage: week or two	Malaise and pallor	Loss of red blood cells
	Infections & and inflammation	Low white cell count and GI damage
	Haemorrhage	Low platelet count
	Erythema	Loss of skin cells
	Epilation (loss of hair)	Damage to skin cells
	Diarrhoea and emaciation	GI tract damage and lack of water re-absorption

It is possible to improve the chances of survival of individuals exposed to doses approximating or greater than the LD_{50/60} by stimulating viable bone marrow stem cells or by substituting new bone marrow or concentrates of bone marrow stem cells from a suitable donor, together with appropriate medical care (fluid replacement, barrier nursing, antibiotics and antifungal drugs).

At doses exceeding about 5 Gy additional effects occur, including severe gastrointestinal (stem cell and endothelial capillary cell) damage which, when combined with bone marrow damage, cause death in one to two weeks.

At about 10 Gy acute inflammation of the lungs can occur, leading to death. At even higher doses effects on the nervous and cardiovascular systems occur and the individual dies of shock after a few days.

Although acute radiation syndrome (ARS) is not common there have been several instances during the last 70 years, mainly as a result of accidents. The most significant of these is the Chernobyl nuclear reactor accident in 1986 where it is believed that there were 28 fatalities from ARS and about 200 cases of ARS survivors.

Table 2.4 - Somatic Effects of Radiation (Acute Early Effects)

Timescale	Whole Body Dose		
	10 Gy	2 - 10 Gy	2 Gy
Comments	Survival improbable	Survival possible	Survival probable
Week 1	Nausea & vomiting in 1-2 hours	Nausea & vomiting in 1-2 hours	Possible nausea and vomiting
	Latent period in some cases	Latent period: no definite symptoms	Latent period: no definite symptoms
Week 2	Diarrhoea Vomiting Nausea Inflammation of mouth and throat	Loss of hair Loss of appetite and general malaise Fever Haemorrhage	Loss of hair Loss of appetite and general malaise Sore throat Haemorrhage
Week 3	Haemorrhage Fever Rapid emaciation Death	Nose bleeds Pallor Diarrhoea Emaciation, Inflammation of mouth and throat	Nose bleeds Pallor Diarrhoea Moderate emaciation
Week 4		Death in 50% of cases probably through secondary infection	Recovery likely in about 3 months unless complicated by poor health or superimposed injuries or infections

The medical care made available on a short timescale after the Chernobyl accident, included bone marrow transplants, blood transfusion, chemotherapy, antibiotics and techniques to prevent infection. On the whole, these do seem to have been effective in limiting the number of fatalities and many lessons were learnt, which have been made available to the medical profession worldwide.

2.6.2 Part Body Radiation Exposure

When only part of the body is irradiated the effects are generally less drastic, as the rest of the body can help make up deficiencies. However, organs such as the gonads, lens of the eye and the skin can suffer major damage. The skin is noteworthy because it can receive a large, localised

dose particularly from beta emitting contamination. The main reaction is reddening of the skin or erythema. There is an initial reddening which occurs after a few hours (due to histamine release) with true reddening occurring a few weeks later. Other effects are scaling, hair loss and sweat glands not functioning (seen in Table 2.5 below).

Table 2.5 - Tissue Reactions Following Skin Irradiation

Dose (Gy)	Effect
< 3	No observed effects
3 - 10	Short term reddening. Duration of reddening increases with dose. Hair loss occurs. Hair loss is generally permanent above 7 Gy
10 - 50	Radiation Burns with healing over period of weeks, possible permanent scarring
>50	Necrosis and ulcers, which requires skin grafting to heal

Skin dose effects were observed among the firefighters involved in the mitigation operations during the Chernobyl accident. 56 people received burns to greater than 1% of their body and the majority of these were due to beta radiation burns. They were mainly due to highly beta active contaminants being entrained on wet clothing and hence being in close contact with the skin.

The thresholds and effects of an acute irradiation of the gonads are given in Table 2.6.

Table 2.6 - Other Tissue Absorbed Dose Thresholds

Tissue and Effect	Absorbed Dose to Tissue
Testes, temporary sterility	0.3 Gy
Testes, permanent sterility	3.5 - 6 Gy
Ovaries, temporary sterility *	1.5 - 6 Gy
Ovaries, permanent sterility *	3 - 8 Gy

*older women are more susceptible

2.6.3 Late Tissue Reactions

The formation of cataracts in the lens of the eye is an example of a long-term tissue reaction. Cataracts are caused because dead cells cannot be removed from the lens of the eye and cause opacity of the lens. The threshold dose (acute, single exposure) for the formation of detectable opacities is 0.5 - 2.0 Gy and for the formation of cataracts is about 5 Gy.

2.7 Stochastic Effects

2.7.1 Types of Stochastic Effects

Stochastic effects may occur following either an acute or a chronic exposure. Two types of stochastic effects are well recognised. The first is the induction of cancer in the exposed individual and the second is hereditary disorders in the progeny of those exposed.

Cancer is a name for a large group of different diseases in which a cell, usually a single stem cell fails to respond to signals from other cells that should have prevented cell division. As a result, the cell and its descendants continue to divide without limit, producing a tumour. The type of cancer depends on the type of stem cell that is affected and the extent to which normal control is lost.

2.7.2 Cancer Induction

It is assumed that there is no threshold for the induction of molecular change at specific DNA sites involved in the initial events that may result in malignant transformation and potentially cancer. Also, the initial events may involve more than one step in which radiation, or any external trigger, is not necessarily the first. At some time after this initial event, a clone of cells with malignant potential may arise, and after further events in the cells or their environment a cancer may develop. In the development of some cancers, these later stages are age dependent.

The probability of a cancer developing is far lower than the probability of the initial event occurring because of the body's defence mechanisms and the probability of failure of the required succeeding changes. The balance of damage and repair in the cell and the existence of subsequent defence mechanisms are therefore considered to influence the shape of the dose response relationship, but not to result in a threshold. For a stochastic effect the probability of the effect occurring, rather than its severity, is regarded as a function of dose without threshold, as shown in Figure 2.5.

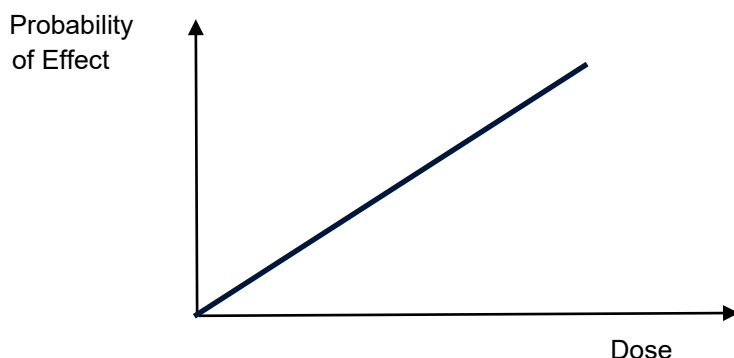


Figure 2.5 - Stochastic effects

In humans, the period between exposure to radiation and recognition of a cancer lasts several years: the latency period. The median latency period may be about eight years in the case of induced leukaemia and two or three times longer in the case of solid tumours. The minimum latency period is the shortest time in which a specified radiation induced tumour is known or believed to occur after exposure. This minimum latency period is two years for acute myeloid leukaemia and about twenty years for other cancers.

2.7.3 Hereditary Stochastic Effects

When the damage caused by radiation occurs in the germ cells, this may be transmitted and lead to hereditary disorders in the descendants of the exposed individual. Radiation has not been identified as a cause of such effects in man, but experimental work, principally studying the mouse, suggests that such effects will occur and that the consequences may range from the undetectably trivial, through gross malformations or loss of function, to premature death. ICRP prudently includes the risk of heritable effects in its system of radiological protection.

Two types of radiation induced genetic damage are considered important: genetic mutation and chromosomal aberration.

Genetic mutation of dominant genes will lead to genetic disease predominantly in the first two generations after exposure. Recessive gene mutations, however, are unlikely to produce any effect in the first few generations but will contribute to genetic damage in future generations.

A general increase in genetic mutations may increase the incidence of multifactorial disorders: conditions known to be due to the interaction of genetic and environmental factors, such as cleft lip and congenital heart defects in children and coronary heart disease in adults. Such an increase in multifactorial disorders has not been observed in man or animals but has been taken into account when assessing the risk factors for hereditary stochastic effects.

Chromosomal aberrations include deletion, duplication or translocation of chromosome segments, or loss or gain of whole chromosomes. A small proportion of chromosomal aberrations will result in congenital abnormalities in the first generation, such as Down's syndrome.

2.8 Radiation Risks - Cancer Induction

Cancer is a common illness; around 50% of people will contract cancer in their lifetime and cancer is responsible for 28% of UK deaths. Only a small fraction of these deaths are radiation induced. Some environmental carcinogens have been recognised, others are unknown.

The consequence of there being many factors that can give rise to cancer and the long latent period is that most cancer deaths are non-attributable. To work out the risks of radiation induced cancer, statistics must be used to compare the death rates to exposed and unexposed populations.

Because the total cancer risk from all causes is so high and the relative risks of exposure to tens of mSv is relatively slight, a very large population (100,000 plus) is needed to identify the increase in cancer mortality. Although this size of study is at present under way (UK National Register of Radiation Workers of over 200,000 people) it is a long-term project due to the significant follow up period that is required due to cancer latency periods. In the meantime, other ways have been used to estimate the risk.

A combination of methods has been chosen:

- follow-up of the 90,000 Hiroshima and Nagasaki atomic bomb survivors (mean dose 0.5 Sv)
- studies of cancer incidence in the radium dial luminising and uranium mining industries
- studies of cancer incidence in patients treated with radiation, e.g. radiotherapy
- animal studies.

None of the above studies are ideal for calculating the risks to radiation workers. The atomic bomb survivors, for example, were exposed to relatively high doses in a very short time, whereas workers receive a much lower dose in a longer time. In addition, the general cancer risks to Japanese and Western people are not the same.

After reviewing all the data, the ICRP initially came up with a cancer risk factor associated with radiation of 1.25% per Sv (ICRP 26 1977).

2.8.1 Changes in Risk Factors

In 1990 the ICRP revised the risk factors associated with radiation exposure and issued the data shown in Table 2.7. Reasons for the increase in risk factors to the current values include:

- increase in the numbers of cancers in the atomic bomb survivors with time
- reduction in the estimated dose to the atomic bomb survivors
- revised modelling of cancer risks
- inclusion of morbidity and hereditary effects into calculation of detriment.

The ICRP also recognised that the risk from a chronic exposure is also less than the risk of an acute exposure (doses of above 200 mSv received at dose rates exceeding 100 mSv^h⁻¹). This is difficult to quantify and hence the statement that the chosen value of 2 for the Dose and Dose rate Effectiveness Factor (DDREF) is somewhat arbitrary but conservative.

The changes in risk factors led to changes in the ICRP recommended dose limits between 1977 (ICRP 26) and 1990 (ICRP 60), i.e. reducing the employee effective dose limit from 50 mSv to 20 mSv.

Table 2.7 - ICRP 60 Excess Lifetime Fatal Cancer Risks

Exposed Group	Excess Lifetime Fatal Cancer Risks (Chronic Exposure) % per Sv	Excess Lifetime Fatal Cancer Risks (Acute Exposure)* % per Sv
Workers	4	8
General Population	5	10

* For doses of above 200 mSv received at dose rates exceeding 100mSv^h⁻¹

2.8.2 Radiation Detriment

In addition to the risk of a fatal cancer, exposure to ionising radiation also gives two additional forms of harm:

- non-fatal cancer
- severe hereditary effects

In recognition of this, ICRP introduced the concept of **radiation detriment**. These detriments are calculated in equivalent terms to fatal cancer as shown in Table 2.8.

Table 2.8 - ICRP 60 Radiation Detriments Associated with Chronic Radiation Exposure

Exposed Group	Detriment (% per Sv)			Total
	Fatal Cancer	Non-Fatal Cancer	Severe Hereditary Effects	
Workers	4.0	0.8	0.8	5.6
General population	5.0	1.0	1.3	7.5

2.8.3 ICRP 103 Risk Coefficients

The ICRP 60 risk coefficients have subsequently been reduced (ICRP 103) and are summarised in Table 2.9. The main reasons for these changes are that:

- the ICRP 103 cancer risk estimates are derived from incidence data whereas the ICRP 60 estimates are based on mortality data, which is now considered to be less reliable; and
- the risk of heritable effects was assessed to be much lower than previously thought.

Table 2.9 - ICRP 103 Detriment-Adjusted Nominal Risk Coefficients Associated with Chronic Radiation Exposure

Exposed Population	Detriment (% per Sv)		
	Cancer Detriment (Weighted)	Hereditary Effects	Total
Adult Workers	4.1	0.1	4.2
Whole Population	5.5	0.2	5.7

Example 2.1

Calculate the individual cancer risk to a member of the public receiving a whole body dose of 3 mSv (use ICRP 103 risk factors)

Solution

Chronic exposure of a member of the public, so use a risk factor of 5.5% per Sv

Excess lifetime cancer risk = $3 \times 10^{-3} \text{ Sv} \times 5.5\% = 1.65 \times 10^{-2} \%$

i.e. lifetime cancer risk has increased from 50% to 50.0165%.

This statistic can also be presented as follows:

$0.0165\% = 1.65 \times 10^{-4}$ which can be interpreted as a risk of 1 in 6060 (i.e. $1/1.65 \times 10^{-4}$)

Example 2.2

Calculate the individual cancer risk to a worker who received an acute whole body dose of 1 Sv over a period of about 10 minutes during an accident.

Solution

Acute exposure of a worker (dose > 200 mSv and dose rate exceeds 100 mSv h^{-1}), so use a risk factor of 8.2% per Sv

Excess cancer risk = $1 \text{ Sv} \times 8.2\% = 8.2\%$

i.e. excess cancer risk of the exposed worker has increased from 50% to 58.2%

2.9 Summary

Components of a cell - nucleus, cytoplasm, cell membrane

Nucleus - contains chromosomes, which are made up of genes

Mitosis - process by which single cells reproduce

Meiosis - a stage in the formation of the reproductive cells (sperm and eggs)

Effects of radiation on cells - inhibition of mitosis, chromosome aberrations, death

Acute effects - effects occurring within a few weeks of a very large exposure, owing to the depletion of cell populations

Chronic effects - effects occurring where the exposure is over a substantial time spans, normally at a low level

Early effects - effects occurring within a short time span (days), only occur with large exposures

Late Effects - effects occurring at later times, typically some years after exposure. The main effect is cancer induction

Somatic effects - damage occurs to the irradiated person

Hereditary effects - damage occurs to the descendants of exposed individuals

Tissue reactions - effects in tissues, the severity of which increases with dose and for which there is a threshold. Mainly early radiation effects plus certain late effects e.g. cataract formation

Stochastic effects - the probability of occurrence depends on dose. Mainly cancer and genetic effects

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

Risk coefficient - the probability of a stochastic effect

2.10 Review Questions

1. Which of the following parts of a mammalian cell is most *radiosensitive*?
 - a. Cell membrane
 - b. Cytoplasm
 - c. Nuclear membrane
 - d. Nucleus
2. Which of the following statements about biological effects of radiation is true?
 - a. Acute exposures do not give rise to stochastic effects
 - b. Lung cancer is an example of a somatic stochastic effect
 - c. The induction of cataracts is a stochastic effect
 - d. Damage to DNA always leads to cell mutation
3. As a result of an accident handling a gamma source, a worker receives a dose of 2 Gy to her hands and 100 mGy to her abdomen. The likely tissue reactions resulting from this exposure are:
 - a. death from radiation sickness within one month
 - b. severe burns to hands but no radiation sickness
 - c. skin reddening to hands and mild radiation sickness
 - d. no short term effects
4. Which of the following best defines a *stochastic* radiation effect?
 - a. No threshold and probability proportional to dose
 - b. No threshold and severity proportional to dose
 - c. A threshold and probability proportional to dose
 - d. A threshold and severity proportional to dose
5. There is a significant delay between irradiation and the formation of a radiation induced cancer because:
 - a. cancer is a tissue reaction and it takes time for the dose to build up above the threshold
 - b. there is a long time between irradiation and cell death for the majority of tissues
 - c. cancers are the result of a series of cellular changes, which take time to occur
 - d. cancer becomes much more likely as the exposed person reaches retirement age
6. The fatal cancer risk coefficient for doses below 200 mGy is:
 - a. equal to the total fatal cancer probability for the Japanese A bomb survivors
 - b. the fatal risk coefficient for high dose / dose rate exposures divided by the DDREF
 - c. derived from animal studies
 - d. the product of the fatal cancer risk coefficient for high dose / dose rate exposures and the DDREF

Chapter 3

Philosophy of Radiation Protection and Legislation

3.1 Aims

By the end of this chapter the student should be able to

- explain the ICRP system of radiation protection - justification, optimisation (ALARP) and dose limitation
- list UK legislation relevant to radiation protection - IRR17, REPP19, EPR16/EASR18 and CDG09 and briefly describe the purpose of each set of regulations
- state the current statutory dose limits (IRR17) for workers and the public.

3.2 The ICRP System of Radiological Protection

The ICRP considered the philosophical basis for radiological protection, based on balancing the harmful effects of radiation with the benefits that can be gained from its use. The ICRP set out the following system of radiological protection.

JUSTIFICATION	No practice involving exposure to ionising radiation should be adopted unless it produces sufficient benefit to the exposed individuals or to society to offset the detriment it causes.
OPTIMISATION	In relation to any particular source within a practice, the individual doses, the number of people exposed and the likelihood of incurring exposure should be kept As Low As Reasonably Achievable (ALARA) ⁴ , economic and social factors being taken into account.
LIMITATION	The exposure of individuals should be subject to dose limits or to some control of risk in the case of potential exposures.

3.2.1 Dose Limitation

Dose limitation when applied to planned occupational exposure has two aims:

- to **prevent** tissue reactions
- to **reduce the risk** of stochastic effects to levels considered **tolerable**.

⁴ In UK law As Low As Reasonably Practicable (**ALARP**) is used in line with the wording of other safety legislation.

3.2.2 Dose Limitation Recommended by the ICRP

The dose limit that has been recommended by the ICRP (ICRP Publication 103, 2007) at a value such that continued exposure at a dose just above the limit would be unacceptable on any reasonable basis. Continued exposure just below the dose limit might be tolerated but would not be welcome, so that acceptable doses are somewhat below the limit. In order to recommend dose limits it was therefore necessary to compare the risks of exposure with the tolerability and acceptance criteria given above.

3.3 UK Legislation

The main pieces of legislation in relation to radiation protection are summarised below:

Ionising Radiations Regulations 2017 (IRR17) - enacted under the Health and Safety at Work Act 1974 and regulated by the Health and Safety Executive (HSE) or the Office of Nuclear Regulation (ONR) on nuclear licensed sites. The aim of the regulations is to protect workers and others from ionising radiation in the workplace.

Radiation Emergency (Preparedness and Public Information) Regulations 2019 (REPIR19) - enacted under the Health and Safety at Work Act 1974 and regulated by the HSE or ONR on nuclear licensed sites. The regulations establish a framework of preparedness measures to ensure that arrangements are in place to effectively respond to that emergency, both on the site of the emergency situation and off-site where members of the public might be affected.

Environmental Permitting Regulations 2016 (EPR16) / Environmental Authorisations (Scotland) Regulation 2018 (EASR18) - EPR16 applies to England and Wales, with the equivalent legislation EASR18 for Scotland. Users of radioactive material are required to hold an environmental permit to use and keep radioactive substances and to accumulate and dispose of radioactive waste. These regulations do not apply in the case of MOD controlled sites, but the Secretary of State for Defence has stated that standards and arrangements will be, so far as is reasonably practicable, at least as good as those required by legislation.

Nuclear licensed sites are exempt from provisions of EPR16 that relate to registration of keeping and use of radioactive material and the accumulation of radioactive waste due to the provision within Licence Conditions. These sites still require a permit to dispose of radioactive material.

In addition to enabling the EPR16, the Environment Act 1995 established the Environment Agency (EA) as the sole regulator for the disposal and discharge of radioactive wastes by permit holders, including the operators of nuclear installations in England and Wales. The Scottish Environment Protection Agency (SEPA) and the Northern Ireland Environmental Protection Agency (NIEPA) fulfil this role in Scotland and Northern Ireland respectively.

Carriage of Dangerous Goods and Use of Transportable Pressure Equipment Regulations 2009 (CDG09) - regulates the carriage of dangerous goods by road and rail in Great Britain including radioactive materials. It is regulated by the ONR.

3.4 Statutory Radiation Dose Limits

The latest ICRP recommended revised dose limits have been incorporated into Regulation 12 & Schedule 3.3 of IRR17. These are summarised in Table 3.1.

Table 3.1 - Annual Dose Limits - Ionising Radiation Regulations 2017

Person	Effective Dose (mSv)	Skin, hands, forearms feet & ankles Equivalent Dose (mSv)	Lens of the eye Equivalent Dose (mSv)
Employee	20	500	20
Trainee under 18	6	150	15
Any other person	1	50	15

3.5 Summary

ICRP System of Radiological Protection - Justification, optimisation and limitation

Aims of dose limitation - to prevent tissue reactions and to reduce the risk of stochastic effects to levels considered tolerable

Dose Limits - contains chromosomes, which are made up of genes

Person	Effective Dose (mSv)	Skin, hands, forearms feet & ankles Equivalent Dose (mSv)	Lens of the eye Equivalent Dose (mSv)
Employee	20	500	20
Trainee under 18	6	150	15
Any other person	1	50	15

Ionising Radiations Regulation 2017 - protection of workers from effects of ionising radiation

Radiation Emergency (Preparedness and Public Information) Regulations 2019 - framework of preparedness measures to ensure that arrangements are in place to effectively respond to that emergency, both on the site of the emergency situation and off-site where members of the public might be affected

Environmental Permitting Regulations 2016 / Environmental Authorisations (Scotland)

Regulation 2018 - users of radioactive material are required to hold an environmental permit to use and keep radioactive substances and to accumulate and dispose of radioactive waste. Do not apply in the case of MOD controlled sites

Carriage of Dangerous Goods and Use of Transportable Pressure Equipment Regulations 2009 - regulates the carriage of dangerous goods by road and rail in Great Britain including radioactive materials

3.6 Review Questions

1. In radiation protection, *justification* can be said to mean
 - a. all doses of radiation must be kept as low as reasonably practicable
 - b. all doses must be kept to a level in proportion to the importance of the work activity
 - c. all practices involving ionising radiation must show a net positive benefit
 - d. the individuals receiving the dose must receive the benefit which results from the practice
2. In order to protect against tissue reactions of radiation, dose limits should be set
 - a. below the threshold for such effects
 - b. at a level such that the risk is tolerable
 - c. to a level equal to the average annual background radiation dose
 - d. such that all doses are kept ALARP
3. The dose limit to a member of the public is
 - a. no specified value
 - b. 20 mSv
 - c. 15 mSv
 - d. 1 mSv

Chapter 4

External Radiation Hazards

4.1 Aims

By the end of this chapter the student should be able to

- estimate radiation dose rates using simple calculational models
- explain the methods of protection associated with time, distance and shielding.

4.2 Radiation Dose rate from a Point Source

The number of particles or photons crossing an area of 1 m² in 1 s is called the fluence rate or flux (ϕ). For a point source, as shown in Figure 4.1, which emits Q photons per second the flux is the number of photons passing through an area of 1 m². Since the photons are being emitted uniformly in all directions, the flux at distance r will be given by:

$$\phi = \frac{Q}{4\pi r^2} \quad 4.1$$

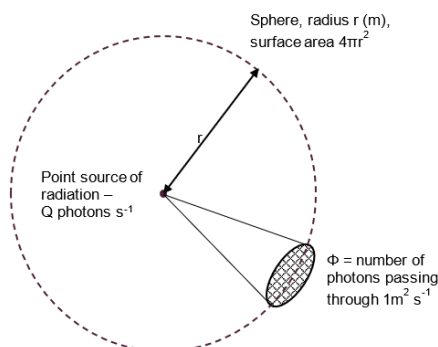


Figure 4.1 - Flux from a point source

The radiation dose rate is proportional to the flux and can therefore be expressed as:

$$D = \frac{\text{constant}}{r^2} \quad 4.2$$

The value of the constant of proportionality will be dependent on the activity, A , of the source and on the properties of the radiation emitted. The radiation dose rate can therefore be expressed as:

$$D = \frac{A\tau}{r^2} \quad 4.3$$

where D is the dose rate in mSv h⁻¹, A is the activity of the source in MBq, τ is the gamma ray constant in mSv h⁻¹ MBq⁻¹ at 1 m and r is the distance from the point source in m.

Examples of values of the gamma ray constant are given in Table 4.1. A more extensive list can be found in <https://www.osti.gov/pages/servlets/purl/1607209>.

Table 4.1 - Gamma Ray Constants

Isotope	Gamma Ray Constant (mSv h ⁻¹ MBq ⁻¹ at 1 m)
⁶⁰ Co	3.062 x 10 ⁻⁴
¹³⁷ Cs / ^{137m} Ba	8.228 x 10 ⁻⁵
¹⁹² Ir	1.167 x 10 ⁻⁴
²⁴¹ Am	5.413 x 10 ⁻⁶

In situations where the gamma ray constant is not known a useful expression for calculating the approximate dose rate from a γ source is:

$$D = \frac{ME}{7r^2} \quad 4.4$$

where D is the dose rate in $\mu\text{Sv h}^{-1}$, M is the activity of the source in MBq, E is the gamma (photon) energy per disintegration in MeV and r is the distance from the source in m. Values for E can be obtained from ICRP 107 Nuclear Decay Data for Dosimetric Calculations.

(<https://www.icrp.org/publication.asp?id=ICRP%20Publication%20107>).

4.2.1 Inverse Square Law

From Equations 4.2 to 4.4 it can be seen that the radiation dose rate from a point source is inversely proportional to the distance from the point source and therefore:

$$D_1 r_1^2 = D_2 r_2^2 \quad 4.5$$

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

In practice it is not always possible to approximate a source to be point source geometry.

However, it may be possible to use a simple geometry such as a line source or a disc source. It can be seen in these situations that the dose rate around the source does not follow the inverse square law.

Example 4.1

Calculate the dose rate at 1 m from a 5 GBq ⁶⁰Co point source using equations 4.3 and 4.4.

What will the dose rate be at 0.2 m.

Solution

At 1 m

Using equation 4.3: $D = 5 \times 10^3 \times 3.062 \times 10^{-4} / 1^2 = \mathbf{1.5 \text{ mSv h}^{-1}}$

Using equation 4.4: $D = 5 \times 10^3 \times 2.5038 / (7 \times 1^2) = \mathbf{1.8 \text{ mSv h}^{-1}}$

At 0.2 m

Using equation 4.3, dose rate at 1 m is 1.5 mSv h⁻¹, so dose rate at 0.2 m is $1.531/0.2^2 = \mathbf{38 \text{ mSv h}^{-1}}$

Using equation 4.4, dose rate at 1 m is 1.8 mSv h⁻¹, so dose rate at 0.2 m is $1.788/0.2^2 = \mathbf{45 \text{ mSv h}^{-1}}$

Example 4.2

Calculate the activity of a ^{22}Na source which gives a dose rate of $6.4 \mu\text{Svh}^{-1}$ at 1 m using equations 4.3 and 4.4.

Solution

Rearranging equation 4.3. gives to give $A = \frac{Dr^2}{\tau}$

so $A = 6.4 \times 10^{-3} \times 1^2 / 2.851 \times 10^{-4} = \mathbf{22 \text{ MBq}}$

Rearranging equation 4.4. gives to give $M = \frac{7Dr^2}{E}$

so $M = 7 \times 6.4 \times 1^2 / 2.1926 = \mathbf{20 \text{ MBq}}$

4.3 Radiation Dose Rate from a Line Source

The dose rate at point P, from a line source, as shown in Figure 4.2 can be calculated using Equation 4.6:

$$D = \frac{A\tau\theta}{Lr} \quad 4.6$$

where D is the dose rate in mSvh^{-1} , A is the activity of the source in MBq, τ is the gamma ray constant in $\text{mSvh}^{-1} \text{MBq}^{-1}$ at 1 m, L is the length in m of the line source, r is the perpendicular distance in m from the line source to point P and θ is the angle, in radians⁵, subtended from point P to the ends of the line source.

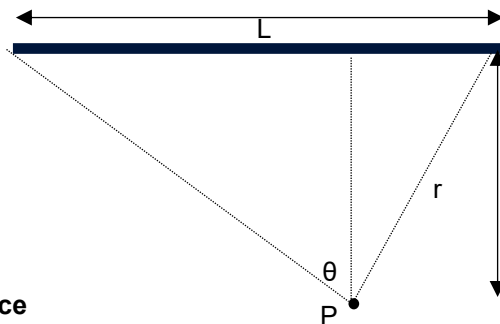
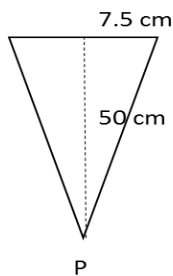


Figure 4.2 - Line Source

Example 4.3

Calculate the dose rate 50 cm from the midpoint of a 15 cm ^{192}Ir line source

Solution



From Equation 4.6 $D = \frac{A\tau\theta}{Lr}$

$\tan(\theta/2) = 7.5/50$ so $\theta = 17^\circ = 17/360 \times 2\pi = 0.30$ radians

So $A = 60 \times 10^9 \text{ Bq}$, $\tau = 1.167 \times 10^{-4} \text{ mSvh}^{-1} \text{MBq}^{-1}$ at 1 m, $\theta = 0.30$ radians, $L = 0.15 \text{ m}$ and $r = 0.5 \text{ m}$.

Substituting in equation 4.6 gives

$D = 60 \times 10^3 \times 1.167 \times 10^{-4} \times 0.30 / (0.15 \times 0.5) = \mathbf{28 \text{ mSvh}^{-1}}$

⁵ 2π radians = 360° , so for example $45^\circ = 45/360 \times 2\pi$ radians = 0.79 radians

4.4 Radiation Dose Rate from a Disc Source

The dose rate at a perpendicular distance from the centre of a disc source can be calculated using Equation 4.7:

$$D = \frac{A\tau}{a^2} \ln \left[1 + \frac{a^2}{r^2} \right] \quad 4.7$$

where D is the dose rate in mSv h⁻¹, A is the activity of the source in MBq, τ is the gamma ray constant in mSv h⁻¹ MBq⁻¹ at 1 m, a is the radius of the disc in m and r is the perpendicular distance in m from the centre of the disc source.

Example 4.4

Calculate the dose rate 75 cm (perpendicular distance) from the centre of a 20 cm radius 120 GBq ¹⁹²Ir disc source.

Solution

From Equation 4.7

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

so A = 120 x 10³ MBq, τ = 1.167 x 10⁻⁴ mSv h⁻¹ MBq⁻¹ at 1m, a = 0.2 m, r = 0.75 m. Substituting in Equation 4.7 gives:

$$D = (120 \times 10^3 \times 1.167 \times 10^{-4} / 0.2^2) \times \ln (1 + 0.2^2 / 0.75^2) = \mathbf{24 \text{ mSv h}^{-1}}$$

4.5 Methods of Protection

4.5.1 Time

The dose received as a result of exposure to a constant dose rate is directly proportional to the duration of the exposure (time), as defined by Equation 4.8:

$$\text{Dose} = \text{Dose rate} \times \text{time} \quad 4.8$$

Example 4.5

An improvement in technique has enabled a valve to be removed in two hours instead of the usual three. If the mean effective dose rate around the valve is 250 μ Sv h⁻¹, what is the dose reduction?

Solution

Original dose	=	3 h $\times$ 250 μ Sv h ⁻¹	=	750 μ Sv
New dose	=	2 h $\times$ 250 μ Sv h ⁻¹	=	500 μ Sv
Dose reduction	=	250 μSv		

Applying the same principle, stay times can also be determined if a dose limit is set.

Example 4.6

A radiation worker works for 50 weeks per year. How many hours per week can he work in an area in which the effective dose rate is $25 \mu\text{Sv h}^{-1}$, assuming (a) that local policy is to limit annual doses to 5 mSv or (b) the legal annual dose limit of 20 mSv applies?

Solution

$$\text{Time} = \frac{\text{Dose}}{\text{Dose rate}}$$

(a) Exposure time to receive a dose of 5 mSv = $5000/25 = 200$ hours = **4 hours per week**

(b) Exposure time to receive a dose of 20 mSv = $20000/25 = 800$ hours = **16 hours per week**

There are various methods that can be used to minimise doses by reducing the time spent in a radiation area:

- **Preplanning:** the time spent carrying out work in a radiation area must be reduced by careful pre-planning and the use of approved procedures.
- **Practice:** teams must be familiar with the plant and have practiced on 'mock-ups', if necessary.
- **Training:** teams must be adequately trained in the hazards of radiation and in the methods of protection.
- **Waiting Areas:** low dose rate waiting areas must be provided to minimise the time a team spends in a radiation area while waiting to commence a task.
- **Work Sharing:** the time an individual spends in a radiation area can be reduced by sharing the task between a number of individuals. This may, however, lead to an increase in the total (or collective) dose for the group of workers concerned.
- **Post Task Review:** Dose estimates: in practice, the time spent in a radiation area is strictly controlled. For a nuclear procedure, an estimate is usually made of the dose that will be received by different groups of workers (e.g. fitters, health physics monitors, contractors). The actual dose received during the procedure can then be compared to the estimate. If the actual dose received is much higher, the Procedure Authorisation Group (PAG) will investigate and discuss the reasons so that lessons can be learned for future procedures. Personnel working in a controlled radiation area may wear a direct-reading (real time display) dosimeter (e.g. a personal electronic dosimeter (PED)) to allow frequent checks on the rate of accumulation of exposure, compilation of statistics and the setting of dose and dose rate alarms.

4.5.2 Distance

From Equation 4.5 it can be seen that increasing the distance from a point source will result in a significant reduction in the dose rate: doubling the distance will reduce the dose rate by a factor of four. For non-point sources, the reduction in dose rate with distance will not be as significant e.g. the dose rate from a line source is inversely proportional to the distance and not the distance squared.

Example 4.7

The dose rate at 2 m from a particular point gamma source is $400 \mu\text{Sv h}^{-1}$. At what distance will the dose rate be $25 \mu\text{Sv h}^{-1}$?

Solution

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

so $400 \mu\text{Sv h}^{-1} \times (2^2) = 25 \mu\text{Sv h}^{-1} \times (r^2)$ and $1600 = 25 r^2$, so $r = 8 \text{ m}$

Example 4.8

Operating procedures require the use of long (1.5 m) handling tools when handling a source in the workplace. The dose rate to the operator when using the tool is $5 \mu\text{Sv h}^{-1}$. Determine the dose rate to the operator if chooses to use a 10 cm long pair of tongs instead

Solution

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

so $5 \mu\text{Sv h}^{-1} \times (1.5^2) = D_2 \times (0.1^2)$, so $D_2 = 1.1 \text{ mSv h}^{-1}$

4.5.3 Shielding

The use of shielding is the preferred method to reduce dose rate, compared with the use of time and distance as it results in intrinsically safer working conditions without the reliance of administrative controls.

The effectiveness of shielding depends on the type of radiation, its energy and the type of shielding material used.

4.5.3.1 Shielding for Alpha Radiation

Alpha radiation is easily absorbed by a thin sheet of paper and hence presents little external hazard. Most alpha particles are stopped by the layer of dead skin cells. However, if they can get in contact with living cells they can cause a lot of damage and hence they are a major internal hazard.

4.5.3.2 Shielding for Beta Radiation

Beta radiation is more penetrating and requires about 10 mm perspex for complete absorption, depending on the energy. When beta particles are slowed down their kinetic energy is turned into X-radiation called bremsstrahlung (braking radiation). The fraction of the incident beta energy converted into photons is dependent on the atomic number of the absorber. For this reason, light materials (low atomic number) such as aluminium or perspex are used for beta shielding.

4.5.3.3 Shielding for Gamma (and X-Ray) Radiation

Gamma, unlike alpha and beta radiation, is never completely stopped, only attenuated exponentially when passing through any material. As most of the gamma interactions in matter are with electrons, materials with high mass numbers (hence high electron densities/ atomic numbers) are chosen for shielding. Typical materials are lead, tungsten and depleted uranium. Other materials may be chosen for cheapness, such as concrete or water.

The dose rate D_t from a collimated beam⁶ passing through a shield is given by:

$$D_t = D_0 e^{-\mu t} \quad 4.9$$

where μ is the linear attenuation coefficient of the shielding material which is a function of the photon energy and the shielding material and has units of m^{-1} , t is the thickness of the shielding material (in m) and D_0 is the initial dose rate and is shown in Figure 4.3.

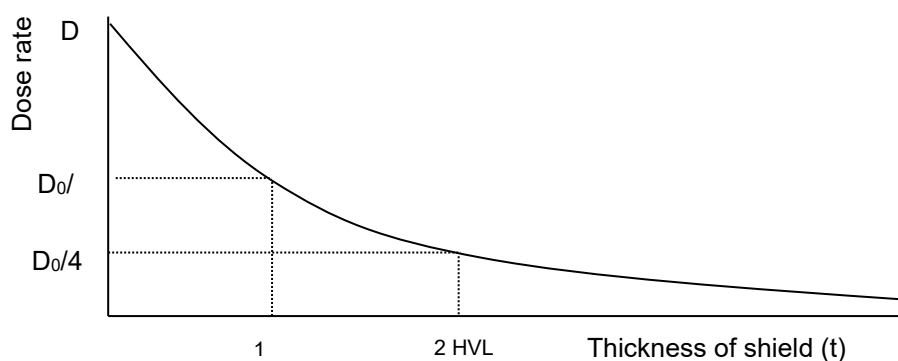


Figure 4.3 - Dose rate of a collimated photon beam passing through a shield

The half value layer (HVL) for a particular shielding material is the thickness required to reduce the dose rate to one half of its incident value and is inversely proportional to the linear absorption coefficient derived below and shown in Equation 4.11:

$$\frac{D_1}{D_0} = 0.5 = e^{-\mu t_{1/2}} \quad 4.10$$

and rearranging gives:

$$HVL = \frac{\ln(2)}{\mu} = \frac{0.693}{\mu} \quad 4.11$$

Examples of half value layers are given in Table 4.2.

Table 4.2 - Typical Half Value layers in Lead and Concrete

Source and Energy	HVL (cm)	
	Lead	Concrete
⁶⁰ Co (1.17 MeV & 1.33 MeV)	1.2	6.6
¹³⁷ Cs (0.662 MeV)	0.7	5.1

⁶ More information on broad beam radiation is covered in Nuclear Physics lectures in this module

Example 4.9

The dose rate near a valve is 0.8 mSv h^{-1} and is due to γ photons from ^{60}Co inside the valve. Work has to be carried out nearby. What thickness of shield is required to reduce the gamma dose rate to 0.1 mSv h^{-1} ?

Solution

The dose rate is to be reduced by a factor of 8, which is equivalent to 3 HVL. The HVL for lead / ^{60}Co is 1.2 cm (Table 4.1), so **3.6 cm** is required.

4.5.3.4 Shielding for Neutron Radiation

Neutrons passing through a shield interact either through scattering or absorption reactions. For effective shielding, maximum absorption is required. Most shield materials, however, absorb only low energy neutrons. Neutron shielding is therefore a two-stage process:

1. Scattering (moderation) to reduce the neutron energy
2. Absorption of the neutron

The ideal neutron shield therefore includes a light material for moderation (e.g. water or polythene) and an effective neutron absorber, for example boron.

It should be noted that it is essential to choose an absorber that will not emit energetic capture gammas on absorbing the neutron.

4.6 Summary

Point Source

Approximate:

$D = \frac{ME}{7r^2}$ where D is the dose rate in $\mu\text{Sv h}^{-1}$, M is the activity of the source in MBq, E is the gamma energy per disintegration in MeV and r is the distance from the source in m.

More accurate:

$D = \frac{A\tau}{r^2}$ where D is the dose rate in mSv h^{-1} , A is the activity of the source in MBq, τ is the gamma ray constant in $\text{mSv h}^{-1} \text{MBq}^{-1}$ at 1m and r is the distance from the point source in m.

Inverse Square Law

$D_1 r_1^2 = D_2 r_2^2$ where D_1 = dose rate at distance r_1 from the source and D_2 = dose rate at distance r_2 from the source

Methods of Protection

Time: dose is proportional to exposure time

Distance: inverse square law applies to point sources

Shielding: beta use low Z e.g. Perspex, aluminium, gamma/X-ray use high Z e.g. lead, tungsten, depleted uranium, neutron use hydrogenous material e.g. water, polythene + neutron absorber e.g. boron

4.7 Review Questions

1. If a worker is due to spend 20 hours per working week in a particular area and his annual dose constraint is 2 mSv what is the maximum permissible average dose rate?
 - a. $2 \mu\text{Sv h}^{-1}$
 - b. $100 \mu\text{Sv h}^{-1}$
 - c. $10 \mu\text{Sv h}^{-1}$
 - d. 2 mSv h^{-1}
2. The dose rate from a particular point source is $500 \mu\text{Sv h}^{-1}$ at 6 m. At what distance will it give a dose rate of $5 \mu\text{Sv h}^{-1}$?
 - a. 1 m
 - b. 600 m
 - c. 60 m
 - d. 10 m
3. A point source gives a dose rate of $20 \mu\text{Sv h}^{-1}$ at 0.5 m. Calculate the dose that a worker would receive if he works at a distance of 1.5 m for 2 hours and then for a further 5 hours at a distance of 3 m.

Chapter 5

Internal Radiation Hazards and Methods of Protection

5.1 Aims

By the end of this chapter the student should be able to

- state the routes of entry of radionuclides into the body
- describe the behaviour of radionuclides in the body
- define the term effective half life, committed effective dose, annual limit of intake
- explain the concept of a dose coefficient and explain the factors influencing their magnitude
- state the probable sources of contamination in the workplace
- describe the methods of protection from internal hazards

5.2 Internal Radiation Hazard

When radioactive material is contained inside a sealed container it may give rise to an external radiation hazard to nearby personnel. Conversely, if the material is not contained it presents a potential internal radiation hazard due to the possibility of some of the material entering the body.

Once the radioactive material is inside a person it will continue to irradiate them until it is excreted or has decayed. Some radionuclides may remain in the body for many years and during this period can give rise to significant absorbed doses. Therefore, quite small quantities of radioactive material, which, if securely contained would represent an insignificant external hazard, do present appreciable internal radiation hazards.

Uncontained radioactive material is generally referred to as radioactive contamination.

5.3 Routes of Entry

Radioactive materials can enter the body in four ways:

1. **Ingestion** of contaminated food, water etc.
2. **Inhalation** of airborne contamination
3. **Injection** through wounds or broken skin
4. **Absorption** through intact skin

Once a radioactive material has entered the body, knowledge of both the physiological characteristics of the person and the material's chemical and physical properties is essential to determine what happens to it. For the purposes of radiological protection the ICRP has published basic anatomical and physiological data for use in radiation protection (ICRP 89), for example reference man has a mass of 73 kg, contains 42 litres of water and excretes 1.6 litres of urine per day.

5.3.1 Ingestion

The ICRP human alimentary tract model (ICRP 100) is used to predict the behaviour of ingested materials by children and adults, providing age-dependent parameter values for the dimensions of the alimentary tract regions, and associated transit times for the movement of materials through these regions. The default assumption is that radionuclide absorption takes place in the small intestine, but the model allows for absorption in other regions and for retention in or on tissues within the alimentary tract when information is available. Doses are calculated to target cells for cancer induction in the oral cavity, oesophagus, stomach, small intestine, and colon.

The main factor for ingestion is the chemical form of the radionuclide, which determines its solubility and hence its ability to cross the gut lining and enter the body fluids. Insoluble materials are estimated to have a stay time within the body of approximately 24 hours, before being excreted in the faeces. Therefore, for insoluble materials, generally ingestion gives rise to lower doses than inhalation.

The fractional absorption of an ingested radionuclide from the human alimentary tract to blood is specified as an f_A value and is generally a single value for all chemical forms of the element as shown in Table 5.1.

Table 5.1 - Classification of Ingested Particulate Matter (ICRP Electronic Occupational Intakes of Radionuclides (OIR) Viewer)

Element	f_A	Compounds
cobalt	0.1	All unspecified compounds
	0.05	Insoluble oxides
iodine	0.99	All unspecified compounds
caesium	0.1	Relatively insoluble forms, irradiated fuel fragments
	0.99	Caesium chloride, nitrate, sulphate; all unspecified compounds
radium	0.2	All compounds
plutonium	0.0005	Soluble forms: nitrate, chloride, bicarbonates, all other unspecified compounds
	0.00001	Insoluble oxides

5.3.2 Inhalation

The ICRP has published a human respiratory tract model (ICRP 66) for reference man. This model is used to predict the behaviour of inhaled gases, vapours and particulates within the respiratory tract. This was revised slightly as described in ICRP 130 (the major change made relates to the clearance of deposited material by both particle transport and absorption into blood), prior to establishing the revised dose coefficients described below.

In the case of particulates, the model gives values of the fraction of the inhaled activity that is deposited in the lung, promptly exhaled, caught in the back of the throat and swallowed and trapped in the nose and blown out. These fractions are largely dependent upon the physical properties of the particle such as its diameter and density. Large particles are likely to be stopped before reaching the lung cavity, whilst very small particles may enter the lung but be promptly exhaled. If the material is deposited in the lung, the respiratory tract model can be used to predict the rate of uptake into the bloodstream. Soluble materials, such as iodine are rapidly transferred through the lung wall, whilst insoluble materials, such as heavy metal oxides, can remain in the lung for many years.

The solubility of particulate materials deposited in the respiratory tract is specified in terms of lung clearance types where

- Type V, very fast absorption rate into the blood
- Type F, fast absorption rate into the blood
- Type M, moderate absorption rate into the blood
- Type S, slow absorption rate into the blood

Recommended default absorption types for the various elements are given in the ICRP Electronic Occupational Intakes of Radionuclides (OIR) Viewer and several are also shown in Table 5.2 where f_A is the fraction of the material that will be absorbed into the bloodstream.

Table 5.2 - Classification of Inhaled Particulate Matter (aerosols unless specified otherwise) (ICRP Electronic (OIR) Viewer)

Element	Type	f_A	Compounds
cobalt	F	0.1	Cobalt nitrate
	M	0.02	Unspecified compounds
	S	0.001	Cobalt oxide
iodine	F	0.9	Sodium iodide, caesium chloride vector, silver iodide, all unspecified forms
	M	0.2	
	S	0.01	
iodine gas or vapour	F	0.99	Elemental iodine and unspecified forms
caesium	F	0.99	Caesium chloride, nitrate, sulphate
	M	0.2	Irradiated fuel fragments, all unspecified forms
	S	0.01	
radium	F	0.2	Nitrate
	M	0.04	All unspecified forms
	S	0.002	
plutonium	F	0.0005	
	M	0.0001	Plutonium citrate, tri-butyl-phosphate & chloride
	S	0.0000	Insoluble oxides
		0.0000	^{239}Pu dioxide, plutonium in mixed oxide
		0.0001	Plutonium nitrate

In the case of inhalation of gases and vapours the fraction of the inhaled activity that remains in the body depends upon its solubility and reactivity. They are assigned one of three classes:

- Class SR-0 (insoluble and non-reactive). Only gives external irradiation from submersion in the cloud of gas and internal irradiation from gas within the respiratory tract, applies to inert gases such as example argon and xenon
- Class SR-1 (soluble and reactive). Absorption into the blood may account for less than 100% of inhaled activity
- Class SR-2 (highly soluble and reactive). Complete and instantaneous absorption of all inhaled activity into the blood

5.3.3 Injection

In the case of injection it is usually assumed when modelling the resultant dose that there is immediate 100% uptake into bodily fluids. For many situations, this is unlikely to be the case and there is bound to be some residual activity around the wound site.

5.3.4 Absorption

A few radionuclides, such as tritium, have the ability to cross through the skin barrier and enter the body. Doses from this pathway for most isotopes are normally small compared to those from other routes of intake. However, for exposure to tritiated water vapour the absorption dose is taken to be one half of the inhalation dose.

5.4 Biological and Effective Half-Lives

In the simplest model, the rate of excretion of any substance from the body is assumed to be a single function exponential, like the rate of radioactive decay. Thus, the effective rate of removal of a radioactive substance from the body is equal to the rate of radioactive decay (rad) plus the rate of biological excretion (bio).

$$\frac{dN_{\text{eff}}}{dt} = \frac{dN_{\text{rad}}}{dt} + \frac{dN_{\text{bio}}}{dt} \quad 5.1$$

Therefore, an effective decay constant (λ_{eff}) can be used to express the rate of removal of a radioactive substance from the body as shown in Equation 5.2.

$$\lambda_{\text{eff}} = \lambda_{\text{rad}} + \lambda_{\text{bio}} \quad 5.2$$

Since $\lambda = \frac{\ln 2}{\text{half life}}$ Equation 5.2 can be written as:

$$\frac{\ln 2}{t_{1/2(\text{eff})}} = \frac{\ln 2}{t_{1/2(\text{rad})}} + \frac{\ln 2}{t_{1/2(\text{bio})}} \quad 5.3$$

where $t_{1/2(\text{eff})}$ is the effective half life, $t_{1/2(\text{rad})}$ is the radioactive half life, and $t_{1/2(\text{bio})}$ is the biological half life.

This can be simplified to Equation 5.4

$$\frac{1}{t_{1/2(\text{eff})}} = \frac{1}{t_{1/2(\text{rad})}} + \frac{1}{t_{1/2(\text{bio})}} \quad 5.4$$

If the values of the biological half life and radioactive half life are significantly different the effective half life will approximate to the smaller of the two values, as shown in Table 5.3.

Table 5.3 - Examples of Effective Half Lives

Isotope	Biological Half-life	Radiological Half life	Effective Half-life
^3H	10 days	12.3 years	9.98 days
^{131}I	138 days	8 days	7.56 days
^{60}Co	120 days	5.3 years	113 days
^{239}Pu	200 years	24000 years	198 years

5.5 Committed Equivalent Dose and Committed Effective Dose

Depending on the effective half life, radionuclides incorporated in the human body may give rise to doses to body tissues for many months or years after the intake. For example, all of the dose following an intake of tritium or ^{131}I will be delivered within a year of the intake, whereas for ^{60}Co approximately one eighth of the intake will remain in the body a year after the intake and for ^{239}Pu only a tiny fraction would have been excreted within a year.

To avoid the problem of having to annually reassess the doses arising from each and every internally deposited radionuclide, the ICRP has stated that the dose resulting from an intake of a radioactive substance should be assessed over 50 years (or up to age 70 year for children). Other values of this assessment or integration period (τ) can be used where appropriate.

The **committed equivalent dose** following an intake of a radionuclide is the total dose expected to be delivered within a specific time period. The committed equivalent dose $H_T(\tau)$ in a tissue or organ T is defined by:

$$H_T(\tau) = \int_{t_0}^{t_0+\tau} \dot{H}_0 e^{-\lambda_{\text{eff}} t} dt \quad 5.5$$

where $\dot{H}_0$ is the initial equivalent dose rate at time t_0 . This is illustrated in Figure 5.1.

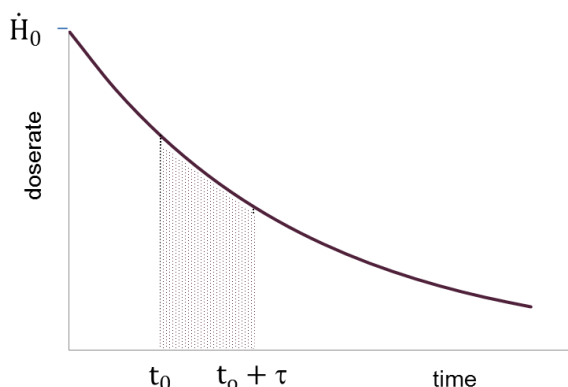


Figure 5.1 - Integration of dose rate to give committed dose (shown by shaded area)

The **committed effective dose** $E(\tau)$ is then given by:

$$E(\tau) = \sum_T w_T H_T(\tau) \quad 5.6$$

The unit for the committed effective dose and the committed equivalent dose is the sievert (Sv).

5.6 ICRP Dose Coefficients

The following factors will affect the dose received following an intake:

- method of entry, (inhalation, ingestion etc.)
 - physical properties of gases, vapours and particulates for inhalation
 - chemical properties, such as solubility for both inhalation and ingestion
- biodistribution
 - chemical properties
- effective half-life
- type and energy of the radiation
 - radiation weighting factor
- daughter nuclides

Models (e.g. respiratory tract, gastrointestinal model) are used to describe the entry of various chemical forms of radionuclides into the body and their distribution and retention after entering the blood. The output of these models is combined with calculations of the energy deposited in target organs to produce dose coefficients for specified radionuclides in terms of SvBq^{-1} .

For workers the inhalation dosimetric data is based on the ICRP 66 human respiratory tract model and the corresponding ingestion data is based on the ICRP 100 human alimentary tract model. The dose coefficients are available in the ICRP OIR series of publications (ICRP 130, 134, 137, 141 and 151) and are available in an electronic database as the electronic OIR viewer.

The corresponding data for members of the public (different age groups) is currently being updated (scheduled for publication during the period 2025-2028) and hence the current dose coefficients can be found in ICRP 119 (based on ICRP 66 and ICRP 30).

ICRP has also made an App available (ICRP Dose Viewer) which can be used to access dose coefficients for members of the public (ICRP119) and for occupational intakes (OIR viewer).

A sample of this data is given in Table 5.4 (workers) and Tables 5.5 and 5.6 (members of the public). For workers, data is given for particle sizes of $1 \mu\text{m}$ and $5 \mu\text{m}$, although the default values should be based on $5 \mu\text{m}$. Data for members of the public is provided for new-borns, 1 year, 5 years, 10 years, 15 years and adults.

The ICRP dose coefficients can be used to compare the relative radiotoxicity of different radionuclides. For example, the committed effective dose per unit intake by inhalation for ^{239}Pu (class M, $5 \mu\text{m}$, worker) is $1.4 \times 10^{-5} \text{ SvBq}^{-1}$ whereas the corresponding value for ^{131}I is $2.7 \times 10^{-9} \text{ SvBq}^{-1}$. This is because

- ^{239}Pu has a longer effective half-life than ^{131}I
- ^{239}Pu is an alpha emitter and ^{131}I is a beta emitter (radiation weighting factor $\alpha = 20$ and $\beta = 1$)

- ^{239}Pu accumulation in the bone therefore exposing the radiosensitive bone marrow whilst iodine accumulates in the thyroid (tissue weighting factor for bone marrow is 0.12 compared to 0.04 for the thyroid)

Table 5.4 - Committed Effective Dose Coefficients for Workers (aerosols unless otherwise specified) (ICRP Electronic OIR Viewer)

Nuclide	Type	Inhalation			Ingestion	
		f_A	e (SvBq ⁻¹) (1 μm)	e (SvBq ⁻¹) (5 μm)	f_A	e (SvBq ⁻¹)
³ H	V - HTO	-	2.0 x 10 ⁻¹¹ (particle size N/A)		0.99	1.9 x 10 ⁻¹¹
	V – HT gas	-	2.0 x 10 ⁻¹⁵ (particle size N/A)		N/A	N/A
⁶⁰ Co	F	0.1	5.0 x 10 ⁻⁹	4.2 x 10 ⁻⁹	0.1	3.2 x 10 ⁻⁹
	M	0.02	1.1 x 10 ⁻⁸	6.2 x 10 ⁻⁹	0.05	2.1 x 10 ⁻⁹
	S	0.001	5.9 x 10 ⁻⁸	3.1 x 10 ⁻⁸		
¹³¹ I	F	0.99	7.2 x 10 ⁻⁹	1.1 x 10 ⁻⁸	0.99	1.6 x 10 ⁻⁸
¹³¹ I (gas or vapour)	F	0.99	1.7 x 10 ⁻⁸			
¹³⁷ Cs	F	0.99	6.0 x 10 ⁻⁹	9.3 x 10 ⁻⁹	0.1	1.6 x 10 ⁻⁹
	M	0.2	7.9 x 10 ⁻⁹	5.6 x 10 ⁻⁹	0.99	1.4 x 10 ⁻⁸
²²⁶ Ra	F	0.2	1.5 x 10 ⁻⁷	1.6 x 10 ⁻⁷	0.2	1.3 x 10 ⁻⁷
	M	0.04	2.1 x 10 ⁻⁶	1.4 x 10 ⁻⁶		
²³⁹ Pu	²³⁹ PuO ₂	0.000002	4.5 x 10 ⁻⁵	2.5 x 10 ⁻⁵	0.0005	2.4 x 10 ⁻⁹
	²³⁹ PuNO ₃	0.0001	2.3 x 10 ⁻⁵	1.3 x 10 ⁻⁵	0.0001	1.2 x 10 ⁻⁷

Table 5.5 - Committed Effective Dose Coefficients for Inhalation of Radionuclides for Members of the Public (ICRP 119 Annex G)

Nuclide	Type	Infant			e (SvBq ⁻¹)		
		f_1	e (SvBq ⁻¹)	f_1	1 year	10 year	Adult
^{60}Co	F	0.6	3.0 x 10 ⁻⁸	0.3*	2.3 x 10 ⁻⁸	8.9 x 10 ⁻⁹	5.2 x 10 ⁻⁹
	M	0.2	4.2 x 10 ⁻⁸	0.1	3.4 x 10 ⁻⁸	1.5 x 10 ⁻⁸	1.0 x 10 ⁻⁸
	S	0.02	9.2 x 10 ⁻⁸	0.01	8.6 x 10 ⁻⁸	4.0 x 10 ⁻⁸	3.1 x 10 ⁻⁸
^{131}I	F	1.0	7.2 x 10 ⁻⁸	1.0	7.2 x 10 ⁻⁸	1.9 x 10 ⁻⁸	7.4 x 10 ⁻⁹
	M	0.2	2.2 x 10 ⁻⁸	0.1	1.5 x 10 ⁻⁸	4.7 x 10 ⁻⁹	2.4 x 10 ⁻⁹
	S	0.02	8.8 x 10 ⁻⁹	0.01	6.2 x 10 ⁻⁹	2.4 x 10 ⁻⁹	1.6 x 10 ⁻⁹
^{137}Cs	F	1.0	8.8 x 10 ⁻⁹	1.0	5.4 x 10 ⁻⁹	3.7 x 10 ⁻⁹	4.6 x 10 ⁻⁹
	M	0.2	3.6 x 10 ⁻⁸	0.1	2.9 x 10 ⁻⁸	1.3 x 10 ⁻⁸	9.7 x 10 ⁻⁹
	S	0.02	1.1 x 10 ⁻⁷	0.01	1.0 x 10 ⁻⁷	4.8 x 10 ⁻⁸	3.9 x 10 ⁻⁸
^{226}Ra	F	0.6	2.6 x 10 ⁻⁶	0.3**	9.4 x 10 ⁻⁷	7.2 x 10 ⁻⁷	3.6 x 10 ⁻⁷
	M	0.2	1.5 x 10 ⁻⁵	0.1	1.1 x 10 ⁻⁵	4.9 x 10 ⁻⁶	3.5 x 10 ⁻⁶
	S	0.02	3.4 x 10 ⁻⁵	0.01	2.9 x 10 ⁻⁵	1.2 x 10 ⁻⁵	9.5 x 10 ⁻⁶
^{239}Pu	F	0.005	2.1 x 10 ⁻⁴	0.0005	2.0 x 10 ⁻⁴	1.2 x 10 ⁻⁴	1.2 x 10 ⁻⁴
	M	0.005	8.0 x 10 ⁻⁵	0.0005	7.7 x 10 ⁻⁵	4.8 x 10 ⁻⁵	5.0 x 10 ⁻⁵
	S	0.0001	4.3 x 10 ⁻⁵	0.00005	3.9 x 10 ⁻⁵	1.9 x 10 ⁻⁵	1.6 x 10 ⁻⁵

* The f_1 value for Type F for an adult is 0.1, **The f_1 value for Type F for an adult is 0.2

Table 5.6 - Committed Effective Dose Coefficients for Ingestion of Radionuclides for Members of the Public (ICRP 119 Annex F)

Nuclide	Infant		f_1	e (SvBq ⁻¹)		
	f_1	e (SvBq ⁻¹)		1 year	10 year	Adult
³ H	1.0	6.4×10^{-11}	1.0	4.8×10^{-11}	2.3×10^{-11}	1.8×10^{-11}
⁶⁰ Co	0.6	5.4×10^{-8}	0.3 [†]	2.7×10^{-8}	1.1×10^{-8}	3.4×10^{-9}
¹³¹ I	1.0	4.8×10^{-7}	1.0	1.8×10^{-7}	5.2×10^{-8}	2.2×10^{-8}
¹³⁷ Cs	1.0	1.1×10^{-8}	1.0	1.2×10^{-8}	1.0×10^{-8}	1.3×10^{-8}
²²⁶ Ra	0.6	5.7×10^{-6}	0.3 ^{††}	9.6×10^{-7}	8.0×10^{-7}	2.8×10^{-7}
²³⁹ Pu	0.005	5.2×10^{-6}	0.0005	4.2×10^{-7}	2.7×10^{-7}	2.5×10^{-7}

[†] The f_1 value for Type F for an adult is 0.1

^{††} The f_1 value for Type F for an adult is 0.1

The ICRP dose coefficients can be used to compare the relative radiotoxicity of different radionuclides.

Example 5.1

The committed effective dose per unit intake by inhalation for ²³⁹PuO₂ (5 µm) is 2.5×10^{-5} SvBq⁻¹ whereas the value for ¹³¹I vapour is 1.1×10^{-8} SvBq⁻¹ (worker data). Explain the difference in these values.

Solution

- ²³⁹Pu has a longer effective half-life than ¹³¹I
- ²³⁹Pu is an alpha emitter whilst ¹³¹I is a beta emitter (radiation weighting factor)
- Plutonium accumulates in the bone therefore exposing the radiosensitive bone-marrow whilst iodine accumulates in the thyroid (tissue weighting factor)

5.7 Calculation of Committed Effective Dose

The committed effective dose is calculated by multiplying the intake by the relevant dose per unit intake value.

$$\text{Committed Dose (Sv)} = \text{Intake (Bq)} \times \text{Dose Coefficient (SvBq}^{-1}\text{)}$$

5.1

Example 5.2

Following an accident, an adult member of the public is assessed to have inhaled 1000 Bq of ¹³¹I (Type M) and 500 Bq of ¹³⁷Cs (Type M). What is the committed effective dose?

Solution

$$\text{Committed Effective Dose} = 1000 \times 2.4 \times 10^{-9} + 500 \times 9.7 \times 10^{-9} = 7.25 \mu\text{Sv}$$

5.8 Annual Limit of Intake (ALI)

Dose coefficients can be used to calculate the value for the limiting intake in one year which when integrated over a period of 50 years will result in a committed effective dose to the whole body of 20 mSv. This intake is called the **Annual Limit of Intake** (ALI). The ALI can be calculated using the equation below:

$$ALI \text{ (Bq)} = \frac{20 \times 10^{-3}}{\text{Dose coefficient (SvBq}^{-1}\text{)}} \quad 5.8$$

Example of dose coefficients and ALIs are given in Table 5.7. It should be noted that the greater the dose coefficient the smaller the ALI and vice versa.

Table 5.7 - Example of Dose Coefficients (5 µm particle inhalation) and ALIs (for a worker)

Isotope	Effective Dose Coefficient (SvBq ⁻¹)	ALI (Bq)
³ H (tritiated water)	2.0 x 10 ⁻¹¹	1.0 GBq
¹³¹ I (Type F vapour)	1.1 x 10 ⁻⁸	1.8 MBq
²³⁹ PuO ₂	1.3 x 10 ⁻⁵	1.5 kBq

5.9 Methods of Protection

A hierarchy of control procedures are used to protect against the internal hazard. The following principles can be applied:

1. minimise as far as possible the activity being handled
2. contain radioactive materials
3. follow correct working procedures, e.g. wearing protective clothing, washing, monitoring etc.
4. follow proper administrative procedures
5. designate areas and set up of barrier procedures
6. routine contamination monitoring

5.9.1 Remove / Minimise Hazard

The first principle requires the use of the radioactive material to be eliminated or reduced as much as possible.

5.9.2 Engineering Controls

The second principle requires the risk of the spread of contamination to be reduced, where possible by engineering or using physical barriers. For very hazardous contamination, work should be carried out in a glove box or remotely using manipulators etc. in a cell. Fume hoods, portable tenting and modular containment structures also enable airborne hazards to be reduced (although fume hoods will involve the discharge of radionuclides to the environment).



(a) Cell line operated using manipulators



(b) Modular containment structure



(c) Glovebox

Figure 5.2 - Examples of Containment Structures

5.9.3 Administrative Controls

Areas where special precautions for contamination are taken need to be designated as controlled areas and change barriers set up.

5.9.3.1 Operating Rules for Contamination Areas

The **local rules** should describe appropriate procedures for contaminated areas. Some rules describe hygiene rules to prevent contamination entering the body:

- no eating, drinking or smoking
- use only disposable tissues
- keep wounds covered up

Other rules will describe appropriate personal protective equipment (PPE) and respiratory protective equipment (RPE). The PPE/RPE which is appropriate will depend on such factors as:

- the level of the internal hazard
- space and time constraints
- temperature and humidity factors
- external dose rates

Selection of the correct RPE requires good judgement. For low levels of airborne contamination a simple filter mask is appropriate. For higher levels, a respirator or, exceptionally, an air-line suit may be required.

5.9.3.2 Contamination Monitoring

Monitoring for contamination is carried out:

- routinely, to confirm the adequacy of equipment and containment in preventing the spread of contamination
- routinely, to confirm that good working practices are being followed
- as required, to assess the hazards involved in a particular task, or to assess the extent of any spills or releases.

Such monitoring can be for surface contamination and/or airborne particulate activity. Surface contamination monitoring can be done directly by probe or indirectly by taking smears and/or wipes.

5.9.3.3 Barrier Procedures

Where protective clothing is to be worn, a change barrier must be set up, with a 'clean' side and a 'dirty' side. No material may cross the barrier without monitoring to check that contamination is not being spread. Contaminated articles must be bagged up (double bagging preferred) prior to crossing the barrier. Care must be taken with tools and monitoring instruments, which may easily become contaminated in the work area. Control barriers may be permanently manned, or exit monitors or other forms of contamination monitoring may be put in place if space suffices.

If a person presents themselves at a barrier and they are contaminated, the Health Physics monitor should:

- reassure the contaminated person
- get them to stand still and keep hands out to their side (to avoid spreading contamination)
- check for the contaminated areas of the body
- arrange for washing of hands and remonitoring if only hands contaminated
- otherwise ask for assistance and move the contaminated person (wrapped in a second coverall) to the prescribed decontamination area to carry out a thorough decontamination

If the person is injured **first aid takes priority over contamination control** and severely injured persons should be transferred to hospital promptly. Where injuries are less serious, some monitoring/decontamination may be appropriate prior to medical treatment.

Contaminated casualties should be escorted by a Health Physics monitor where possible to provide guidance and reassurance to medical personnel who may be unfamiliar with radioactive contamination.

Control barriers may be permanently manned, or else exit monitors or other forms of contamination monitoring may be put in place if space and money suffice.

5.10 Air Monitoring

Air monitoring for particulates can be performed using installed or portable air samplers or small, battery powered personal air samplers. In all cases air is sucked through a filter paper, which is

then either counted in situ by a detector that forms part of the monitor or is removed and counted separately. A portable air sampler is shown in Figure 5.3.



Figure 5.3 - Portable Air Sampler



Figure 5.4 - Overhoff 400 tritium in air monitor

When using a portable air sampler the filter paper is removed after a set time, e.g. shift or day and counted. Knowing the flow rate and the sampling time, the concentration of airborne activity can be calculated using Equation 5.9.

$$\text{Contamination Level (Bqm}^{-3}\text{)} = C_c \times \frac{100}{E_c} \times \frac{1}{V} \quad 5.9$$

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

(Useful conversion factors: $1000 \text{ cm}^3 = 1 \text{ litre}$ and $1000 \text{ litres} = 1 \text{ m}^3$)

Example 5.3

An air sampler running at 100 litres per minute is used to collect an air sample for 5 minutes. This is counted in a beta castle and the gross count rate is 11 cps. Given that the background count rate is 0.5 cps and the counter has an efficiency of 15% calculate the airborne contamination level in Bqm^{-3} .

Solution

Corrected count rate = $11 - 0.5 = 10.5 \text{ cps}$

Volume sampled = $100 \times 5 = 500 \text{ litres} = 0.5 \text{ m}^3$

Airborne Contamination level = $10.5 \times 100/15 \times 1/0.5 = \mathbf{140 \text{ Bqm}^{-3}}$

Specialist equipment can be used to monitor for tritium in air. Examples of these are the Scintrex 209 and the Overhoff 400 (shown in Figure 5.4). These generally report directly in MBqm^{-3} .

5.11 Summary

Routes of Entry

Inhalation

Ingestion

Absorption through Skin

Wounds / injection

Effective Half Life is a combination of the radiological half life and the biological half life where

$$\frac{1}{t_{1/2(\text{eff})}} = \frac{1}{t_{1/2(\text{rad})}} + \frac{1}{t_{1/2(\text{bio})}}$$

Committed Effective Dose –the effective dose received over 50 years, for an adult, or until age 70 for a child, following an intake of radioactive material

$$\text{Committed Effective Dose (Sv)} = \text{Intake (Bq)} \times \text{Dose Coefficient (SvBq}^{-1}\text{)}$$

Annual Limit of Intake

$$\text{ALI (Bq)} = \frac{20 \times 10^{-3}}{\text{Dose coefficient (SvBq}^{-1}\text{)}}$$

Methods of Protection

Remove / Minimise Hazard

Engineering Controls – containment structures – glove boxes, moducons, cells

Administrative Controls – procedures, designated areas, PPE/RPE, barrier arrangements, monitoring

Airborne Monitoring (Particulate)

$$\text{Contamination Level (Bqm}^{-3}\text{)} = \text{Cc} \times \frac{100}{\text{E}_c} \times \frac{1}{\text{V}}$$

5.12 Review Questions

1. For a certain nuclide, the biological half life is 10 days and the radiological half life is 10 days. What is the effective half life of the nuclide?
 - a. 100 days
 - b. 10 years
 - c. 20 days
 - d. 5 days
2. The effective dose coefficient for a form of Pu is $6.7 \times 10^{-5} \text{ SvBq}^{-1}$. The ALI for this form of Pu will be about:
 - a. 300 mSv
 - b. 150 Bq
 - c. 300 Bq
 - d. 20 mSv
3. For a certain nuclide, the biological half life is 500 years and the radiological half life is 2 years. What is the approximate value of the effective half life of the nuclide?
 - a. 500 years
 - b. 251 years
 - c. 2 years
 - d. 1 year
4. A worker has an intake of 500 Bq of ^{60}Co . The relevant effective dose coefficient is $9.6 \times 10^{-9} \text{ SvBq}^{-1}$. The committed effective dose as a result of this intake is:
 - a. 4.8 mSv
 - b. 4.8 μSv
 - c. 1.9 mSv
 - d. 1.9 μSv
5. Which of the following is not likely to lead to contamination?
 - a. Weld radiography with a 200 GBq ^{60}Co source
 - b. Movement of radioactive waste from the work area to a storage point
 - c. Sampling radioactive liquid waste
 - d. Cutting through a contaminated pipe
6. An air sampler is run in a controlled area. After 20 minutes, the filter paper is counted and the results show no counts above background. This shows that, in the controlled area there is no
 - a. detectable airborne activity
 - b. loose contamination
 - c. radiation dose rate above background
 - d. detectable airborne particulate activity

Chapter 6

Designation of Personnel

6.1 Aims

By the end of this chapter the student should be able to

- define the term 'classified person' and state the legal requirements for classified persons
- define the term 'outside worker' and state the legal requirements for outside workers.

6.2 Designation of Personnel

6.2.1 Classified Persons

In accordance with IRR17, the employer should designate as a classified person any employee who is likely to receive an effective dose exceeding 6 mSv or three tenths of any dose limit. The employer must take into account the potential for radiation exposure, including accident situations. Furthermore, he should designate as a classified person any employee who works with radiation sources that it is reasonably foreseeable for the employee to receive an effective dose of 20 mSv or any other dose limit within several minutes.

An employee who becomes a classified person must be informed in writing.

An employee may not become a classified person unless:

- he is over 18
- has been certified fit to work with ionising radiation by the appointed doctor
- has received suitable information, instruction and training.

In deciding whether to classify a person, the employer must consider such factors as the maximum dose rate to which the employee could be exposed, the possibility for accidental exposure and whether the employee will carry out a substantial amount of work in controlled areas.

An individual's dose history should be taken into account when considering classification, but it is not sufficient to deem classification unnecessary based on past doses.

In general, classified persons should remain as such for a calendar year. Exceptions to this are where:

- the appointed doctor requires the person be declassified
- the person ceases to be an employee

- the person moves into a role where it is unlikely he/she will be exposed to an effective dose exceeding 1 mSv per year.

6.2.2 Outside Workers

An outside worker is defined in IRR17 as being any person who carries out services in a controlled or supervised area (see Chapter 8) of an employer other than his own.

Regulation of outside workers was introduced in 1993, to deal with the large numbers of contractors who were being employed within the nuclear industry. The aim of the regulations was to ensure that outside workers were given the same level of supervision and protection as other persons.

Persons who are based at one site and visit other sites owned by the same company are not classed as outside workers when visiting such sites.

Employers of outside workers must find out from the employer in control of the work area what risks and additional training needs are associated with the services the outside worker will perform, well before those services are due to start. This is to ensure that the right workers are chosen to carry out the services and that, before they start, those workers are given the relevant information and training so that they can protect themselves and others properly. Employers in control of the work area also need information about the outside worker (e.g. dose history and training needs) before they start work on site. In most cases, both employers should generally consult their radiation protection advisers (RPAs) on these matters.

Employers of outside workers will require the following information:

- the detail of the actual work to be carried out
- the type of likely radiation exposure
- an estimate of the total dose likely to arise from the work
- the work procedures that will be required to keep doses as low as reasonably practicable (including any use of special protective equipment)
- the risks associated with the work and the precautions that should be taken
- any local restrictions that will be applied
- the local rules that apply in the other employer's site (including emergency arrangements and contingency plans)
- radiation protection supervisor (RPS) appointments
- any relevant dose constraints and associated local investigation levels.

It is the duty of the employer (IRR17 Reg 22(5)) to ensure that each classified outside worker he employs is issued with a radiation passbook, containing the details set out in Schedule 5 of the IRRs. The passbooks are produced by the HSE and issued by the Approved Dosimetry Service (ADS). The employer must ensure that the radiation passbook is kept up to date. Classified outside workers must present their radiation passbook when entering controlled areas of another employer.

6.2.3 Non-Classified Persons

An individual who has not been designated as a classified person is deemed to be a non-classified person.

If working in a controlled area, non-classified persons must work under written arrangements contained within the local rules (see Section 8.2.3.1) for that area. The purpose of the written arrangements is to set out working practices which must be observed to ensure doses above 6 mSv are avoided.

6.3 Summary

Classified Persons

Likely to exceed an effective dose of 6 mSv, or three tenths of any dose limit (legal (IRR17) term)

A non-classified person is someone who has not been designated as a classified person

Must

- Be notified in writing
- Be aged over 18
- Have been certified fit to work with ionising radiation by an appointed doctor
- Have received suitable information, instruction and training

Outside Workers

Any person who carries out services in a supervised or controlled area of an employer other than his own. The employers of outside workers and employers in control of work areas must co-operate to ensure the required level of protection of the outside worker. Classified outside workers must present a radiation passbook when entering controlled areas of other employers.

6.4 Review Questions

1. A person should be designated a classified person if he
 - a. has exceeded a dose limit in the past five years
 - b. has exceeded 6 mSv in the past five years
 - c. works in a controlled area under a written system of work
 - d. is likely to exceed three tenths of any relevant dose limit
2. Which of the following would be deemed a *classified outside worker* in accordance with IRR17?
 - a. An MOD employee who is a non-classified person working within an AWE controlled area
 - b. An MOD classified person working within an AWE controlled area
 - c. An AWE classified person working within an AWE controlled area
 - d. An MOD classified person working within an AWE supervised area

Chapter 7

Dose Assessment and Recording

7.1 Aims

By the end of this chapter the student should be able to

- describe the functions of personal dosimetry
- describe the thermoluminescent dosimeter and its uses and limitations
- describe the electronic personal dosimeter and its uses and limitations
- describe the neutron dosimeter and explain its uses and limitations.

7.2 Introduction

The employer is required in accordance with IRR17 to make an assessment of all significant doses received by employees who are classified persons and to record such dose assessments.

To facilitate this, the employer must make arrangements for dose assessments to be carried out by an ADS, either in house, or under contract. The ADS is approved by the HSE, who require the ADS to demonstrate that suitable quality control procedures are in place.

Records of doses to classified persons must be kept by an ADS for coordination and formal record keeping until the employee reaches 75 years, but in any event for at least 30 years from when they were made. The ADS for coordination and formal record keeping does not need to be the same ADS that is used for dose assessments. The ADS for record keeping has duties to provide employers with dose information within specified time periods.

A list of ADS services in the UK is available from the HSE website (<https://www.hse.gov.uk/radiation/ionising/dosimetry/services-list.htm>), the main providers in the defence industry being the Defence Radiation Protection Service (DRPS) (MOD is the main customer), AWE plc (Atomic Weapons Establishment), Cavendish Nuclear and the Devonport Dosimetry Service.

The form of dose assessment will depend on the work being undertaken. It is the responsibility of the employer to ensure that all significant doses are assessed. For most workers, external dose is the main contributor to the total dose and thermoluminescent dosimeters (TLDs) or PEDs are appropriate. For persons who may receive extremity exposures, extremity dosimetry, such as finger TLDs or lens of the eye TLDs may also be required.

For persons potentially exposed to internal contamination, it is necessary to use methods such as personal air sampling, bioassay (urine and/or faecal sampling) or whole body monitoring to assess committed effective doses.

The employer has the duty to provide (with reasonable notice) a classified person with a copy of his dose record pertaining to the two years prior to the request.

7.3 Personal Dosimetry

Personal dosimetry can be used to

- achieve statutory compliance for classified persons
- highlight poor working practice
- provide a method of dose monitoring within the workplace to backup dose-rate surveys
- gather statistics for future studies
- reassure the workforce
- provide dose information in case of litigation
- provide accident dosimetry for dose assessment of potentially irradiated personnel.

7.4 Thermoluminescent Dosimeters

TLDs are generally used to assess whole body dose (effective dose) and equivalent dose to the skin (as shown in Figure 7.4). Specific TLDs can also be used to assess dose to the lens of the eye or equivalent dose to the extremities (finger TLDs). Most ADSs, such as AWE and DRPS use TLDs as their approved dosimeter.

TLDs consist of crystalline material which can store the energy deposited when it is exposed to different forms of ionising radiation (Figure 7.1). This is achieved by the presence of metastable traps within the crystal structure. The stored energy is released from the material, in the form of light, when it is heated (Figure 7.2). The amount of light released is measured and is proportional to the original energy deposited in the material by the ionising radiation and hence is proportional to the radiation dose received by the individual.

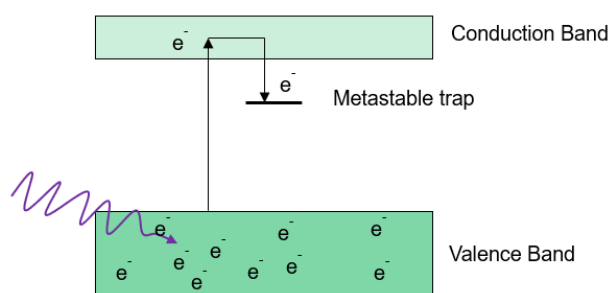


Figure 7.1 - TLD in use

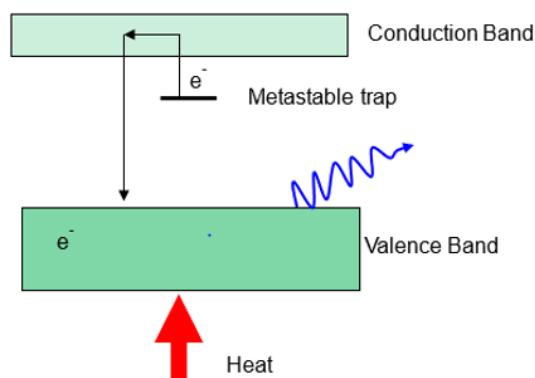


Figure 7.2 - TLD being read out

7.4.1 TLD Read Out

The TLD is read using specially designed equipment. A precise three stage heating cycle is performed using an infra-red lamp. The objectives of this heating cycle are outlined below:

1. **Preheat:** This is done to remove spurious energy depositions
2. **Read Out:** All of the light emitted during this heating cycle is measured and is taken to be proportional to the original energy deposited
3. **Anneal:** This removes any residual energy stored in the thermoluminescent material so that it can be used again.

A typical heating cycle is shown below in Figure 7.3. The light curve produced is usually referred to as the **glow curve**.

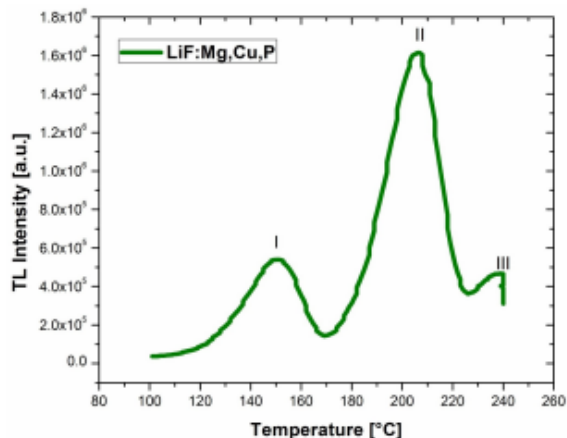


Figure 7.3 - A typical glow curve



Figure 7.4 – DRPS TLD

7.5 Personal Electronic Dosemeters

PEDs as shown in Figure 7.5 can also be used to measure whole body dose. Some electronic dosimeters are also designed to be able to measure the equivalent dose to the skin. More recent developments include the use of filters can be used to give a neutron response. Most of the civil nuclear industry uses electronic dosimeters as their approved dosimeter system (mainly using the Thermo Electronic Personal Dosimeter (EPD)).



Mirion DMC 3000 PED



Thermo Mk 2 EPD (being replaced by TruDose EPD)

Figure 7.5 - Examples of PEDs

Table 7.1 - Advantages and Disadvantages of TLDs and PEDs

	Advantages	Disadvantages
TLD	Thermoluminescent material can be tissue equivalent reusable after annealing linear energy response rapid assessment is possible measure a wide dose range small / not bulky easy to use intrinsically safe	no instant dose readout no permanent record, although the record of the glow curve can be retained does not distinguish between different types of radiation
PED	real time readout. automatic readout into computer files dose and dose rate alarms can record the dose profile can be used as a legal dosimeter	bulk (modern instruments are much better in this respect) sensitivity to moisture sensitivity to radiofrequency interference may be misused as a dose rate meter most designs not HSE approved not intrinsically safe

7.6 Neutron Dosimetry

It has always been problematic trying to determine an accurate equivalent dose for neutrons using some form of personal dosimeter for several reasons:

- the radiation weighting factor (W_R) for neutrons is energy dependent
- the gamma flux associated with neutron flux may interfere with the readout, depending on the material used
- it has proven difficult to identify a material which will be easy to read and will give characteristic responses for the different neutron energies

Stand-alone neutron dosimeters generally are made of CR39 plastic, which is produced by the polymerisation of allyl diglycol carbonate (PADC). When CR39 is subjected to a neutron flux the neutrons undergo elastic scattering reactions with the hydrogen nuclei in the plastic. The resulting recoil protons produce tracks which can be analysed by a combination of a processing technique and an image analysis system.

7.7 Summary

Dose Assessment and Recording - legal requirement for classified persons. Must use an Approved Dosimetry Service (ADS). Records to be kept until the employee reaches age 75 and at least 30 years from when the record was made.

Dosimetry

Whole body beta/gamma dosimetry - TLDs, PED

Neutron dosimetry - CR39 track etch dosimeter

7.8 Review Questions

1. Which of the following is **not** a function of personal dosimetry?
 - a. To highlight poor working practice
 - b. To comply with requirements of the IRRs with respect to classified persons
 - c. To provide reassurance for the workforce
 - d. To provide a shield for radiosensitive organs, when worn near the gonads
2. Which of the following dosimeters does **not** give real-time dose readout?
 - a. AWE TLD
 - b. Thermo EPD
 - c. PD-2i SAIC PED
 - d. Rad 80 PED

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Chapter 8

Designation of Areas

8.1 Aims

By the end of this chapter the student should be able to

- define controlled and supervised areas
- state IRR17 requirements for demarcation, signs, access controls and local rules
- explain the role of the radiation protection supervisor and radiation protection adviser with respect to designated areas.

8.2 Designation of Areas

To control radiation hazards in the workplace, it is important to designate areas appropriately. Areas where there are significant radiological hazards may be designated according to dose rate, level of contamination or the need for special procedures to control dose. The criteria for designation and rules for such areas are set out in IRR17 and summarised below.

8.2.1 Controlled Areas

A controlled area is defined as an area under the control of an employer:

- in which it is necessary to follow special procedures in order to restrict significant exposure to radiation **or**
- where any person working in the area is likely to exceed an effective dose of 6 mSv or a dose in excess of three tenths of any relevant dose limit.

Examples of situations when it is necessary to follow special procedures and hence designate a controlled area are when:

- the external dose rate exceeds $7.5 \mu\text{Sv h}^{-1}$ averaged over a working day
- the dose rate to the hands exceeds $75 \mu\text{Sv h}^{-1}$ averaged over a working day
- there is a significant risk of spreading contamination outside the working area
- it is necessary to prevent or supervise access to the area
- the instantaneous dose rate exceeds $100 \mu\text{Sv h}^{-1}$.

In addition, a controlled area should be designated if the dose rate (averaged over a minute) exceeds $7.5 \mu\text{Sv h}^{-1}$ and site radiography is being undertaken.

The employer in control of the area is responsible for the designation of the area if required, having consulted with an RPA.

Controlled areas do **not** normally need to be set up where:

- work is routine and special precautions are not required e.g. work in the vicinity of a fixed radiation gauge
- work is carried out with low activities of radionuclides of low toxicity inside ventilated enclosures such as fume hoods, or where routine laboratory procedures such as spill trays, goggles and gloves provide adequate protection
- where areas containing sources cannot be physically entered.

8.2.2 Supervised Areas

An employer shall designate as a supervised area any area under his control, not being a controlled area:

- where it is necessary to keep the conditions of the area under review to determine whether it should be designated a controlled area
- where a person is likely to exceed 1 mSv per year or an equivalent dose greater than 5 mSv a year for the lens of the eye or greater than 50 mSv a year for the skin or extremities.

The decision to set up a supervised area depends upon both the likely doses in the area and the probability of conditions changing.

8.2.3 Requirements for Designated Areas

8.2.3.1 Local Rules

The employer must set out local rules in writing wherever a controlled area has been set up.

Local rules may also be required for some supervised areas, depending on the nature of the work and associated hazards.

Local rules serve to identify the key working instructions required to restrict radiation exposure. They can vary in format and can be in the form of instructions, booklets or pictorial posters.

8.2.3.2 Demarcation of Designated Areas

The employer must ensure that any designated area is physically demarcated or, where this is not practicable, delineated by other suitable means. The main purpose of physically demarcating a designated area is to restrict unauthorised access. The suitability type of demarcation for a given area is related to the likelihood of that form of demarcation being effective in access control. Where portable barriers are used, it will normally be necessary to supervise the barrier. It is not necessary to barrier areas which are not normally accessible, e.g. outside first floor windows.

The employer must also display sufficient and suitable signs to indicate that the area is a controlled area and advise on the nature of the radiation and the risks arising from such sources. Example signs are shown in Figure 8.1.

Suitable signs may or may not be required for supervised areas e.g. where the extent of a supervised area is clearly defined in local rules and the extent of the area is well understood by those that work there it might not be appropriate to provide warning signs.



Figure 8.1 - Area Designation Signs

8.2.3.3 Radiation Protection Supervisors (RPS)

An RPS must be appointed for any area subject to local rules. The details of the RPS must be provided in the local rules.

8.2.3.4 Persons who may enter a Controlled Area

The employer may only allow the following types of person to enter a controlled area:

- classified employees
- non-classified persons under a suitable written arrangement
- classified outside workers who have presented an up to date radiation passbook
- patients undergoing medical diagnosis/treatment or their comforters & carers.

All persons entering controlled areas should be appropriately trained.

The written arrangements must ensure that a worker does not receive a dose that would require him to be a classified person and that any other person does not receive a dose in any calendar year in excess of any relevant dose limit.

8.2.3.5 Monitoring of Designated Areas

The employer must ensure that levels of ionising radiation within designated areas are adequately monitored. The purpose of area monitoring is to:

- check that areas have been correctly designated
- determine radiation/contamination levels so that appropriate dose control measures can be suggested
- detect breakdowns in controls or systems
- provide information to support dose assessment made from personal dosimetry or to enable dose assessment where personal dosimetry has failed.

8.3 Summary

Designation of Areas

Controlled (generally greater than $7.5 \mu\text{Sv h}^{-1}$) and supervised

Can be for radiation (external hazard) or contamination (internal hazard)

Requirements: local rules, demarcation, monitoring, access (only classified persons or others under written arrangements)

8.4 Review Questions

1. The legal responsibility for designating areas resides with the:
 - a. HSE
 - b. RPA
 - c. RPS
 - d. employer
2. A chemistry laboratory is used for work with non-radioactive substances. Within the laboratory is a spectrometer. Inside the spectrometer is a high activity sealed beta source which, if handled, can produce a contact dose of about 1 mSv h^{-1} but produces no detectable dose rate at the surface of the instrument. The spectrometer can only be opened using specialist tools.

The laboratory should be:

- a. undesignated
- b. designated a supervised area
- c. designated a controlled area
- d. designated a restricted area

Chapter 9

Environmental Protection

9.1 Aims

By the end of this chapter the student should be able to

- describe the regulatory framework for controlling radioactive discharges in the UK, identifying key legislation and regulatory bodies
- explain the terms BPM and BAT
- identify the main radiation exposure pathways from atmospheric and marine dispersion
- describe the Gaussian Plume Model used for modelling atmospheric releases and explain its limitations
- describe atmospheric discharge deposition mechanisms.

9.2 Regulatory Framework

The requirements of the EU Basic Safety Standards Directive in relation to the control of radioactive material, including radioactive discharges, were originally implemented in the UK within the Radioactive Substances Act 1993 (RSA93). RSA93 has been repealed in England and Wales and its requirements subsumed into Schedule 23 of the Environmental Permitting Regulations 2016 (EPR16). RSA93 has more recently been repealed in Scotland in parallel with the enactment of the Environmental Authorisations (Scotland) Regulations 2018 (EASR18).

9.3 BAT and BPM

Permits to dispose of radioactive wastes require the operator to keep all exposures to the public ALARA, having regard to relevant factors such as protection of the environment and other social or economic impacts – the ‘optimisation requirement’. In England and Wales the application of Best Available Techniques (BAT) is the means to demonstrate compliance with the optimisation requirement. This has replaced the previous requirement to employ Best Practicable Means (BPM). The use of BPM continues to be required by SEPA and NIEPA.

On a practical level, the EA and SEPA consider that the requirements to use BPM are equivalent to the requirements to use BAT and that the obligations on waste producers are the same.

9.3.1 Best Available Techniques (BAT)

In broad terms BAT means the latest stage of development of processes, facilities or methods of

operation which is practicable and suitable to limit waste arisings and disposals. BAT applies throughout the lifetime of a process, from design to implementation, operation, maintenance and decommissioning.

Identification and implementation of BAT implies a balanced judgement of the benefit derived from a measure and the cost or effort of its introduction. The level of effort expended to resolve an issue, and to record the selection process, should be proportional to the scale of the challenge, the range of options available and the extent to which established good practice can be used to assist in the decision making process. Nonetheless, guidance and precedent make clear that practicable measures to further reduce health, safety and environmental impacts can be ruled out as not reasonable only if the money, time, trouble or other costs involved would be 'grossly disproportionate' to the benefit. The following principles should also be taken into account:

- sustainable development
- waste hierarchy – prevent, minimise, reuse, recycle, dispose
- waste form – physical (solid, liquid, gas) and chemical
- the precautionary principle
- the proximity principle – wastes should be treated / disposed of as close to the point that they are generated as possible.

9.3.2 Best Practicable Means (BPM)

BPM for radioactive waste management represents the 'level of management and engineering control that minimises, as far as practicable, the release of radioactivity to the environment whilst taking account of a wider range of factors, including cost-effectiveness, technological status, operational safety, and social and environmental factors'.

BPM requires operators to take all reasonably practicable measures in the design, operation and management of their facilities to minimise disposals of radioactive waste, so as to achieve a high standard of protection for the public and the environment. BPM applies to minimising waste creation, abating discharges, and monitoring of plant, discharges and the environment. It takes account of the availability and cost of measures, operator safety and the benefits to be gained.

9.4 Gaseous Discharges into the Atmosphere

9.4.1 The Gaussian Plume Model

In the Gaussian Plume Model (GPM), the time-averaged concentration profiles in the horizontal (crosswind) and vertical directions are approximated by well-behaved Gaussian (i.e. Normal) distribution, as shown in Figure 9.1.

Mathematically, this can be written:

$$C(x, y, z) = \frac{\dot{Q}}{2\pi u \sigma_y \sigma_z} \exp \left[-\frac{y^2}{2\sigma_y^2} \right] \exp \left[-\frac{z^2}{2\sigma_z^2} \right] \quad 9.1$$

where x is the downwind distance (m), y is the cross-wind distance (m), z is the vertical distance measured from the plume centre-line (m), $\dot{Q}$ is the release rate (Bqs^{-1}), u is the windspeed (ms^{-1}) and σ_y , σ_z are the cross-wind, vertical standard deviations of the plume profile.

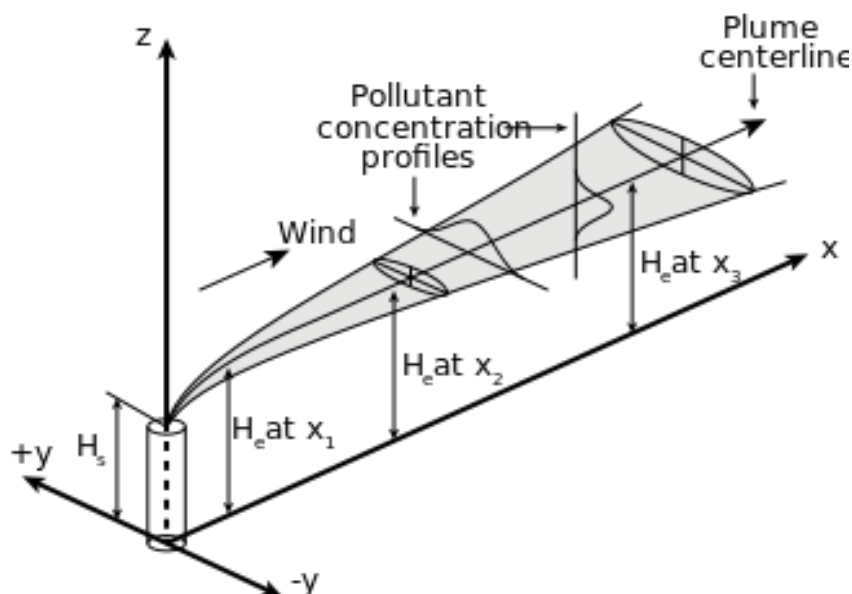


Figure 9.1 - Gaussian Plume Model

Equation 9.1 illustrates some key points about the GPM:

- The concentration is directly proportional to the release rate, i.e. doubling the release rate will cause the concentration to double
- The concentration is inversely proportional to windspeed, i.e. doubling the windspeed will give half the concentration
- The centreline concentration decreases with increasing distance downwind as the dispersion parameters σ_y and σ_z increase.

9.4.1.1 Limitations of the GPM

- It is unlikely that the material will be transported in a straight line, especially at low wind speeds
- The model breaks down at low wind speeds due to the dependence on the model on $1/u$
- No consideration is made of the effect of terrain
- A steady state atmosphere is assumed.

9.4.2 Atmospheric Discharge Deposition Mechanisms

Airborne radioactive material will be deposited on the ground either by dry deposition or wet deposition (i.e. by rainfall).

9.4.2.2 Dry Deposition

The dry deposition rate depends on a large number of factors, the most important being particle size (for aerosols), chemical form (for gases and vapours) and the nature of the surface.

The concentration of activity deposited on the ground, G , is found to be directly proportional to the time-integrated activity concentration immediately above the surface. The constant of proportionality is called the deposition velocity, V_d .

Hence:

$$\text{Deposited Concentration (Bqm}^{-2}\text{)} = \text{TIAC (Bqsm}^{-3}\text{)} \times \text{Deposition velocity (ms}^{-1}\text{)} \quad 9.2$$

The value of the deposition velocity will depend on the precise nature of the discharge and the surface it is depositing on. A great deal of experimental work has been undertaken to measure deposition velocities for various types of discharge/surface. Some of this work has involved the use of wind-tunnels to simulate atmospheric conditions. For radioactive releases in particular, further information has been obtained from studying the fallout from the Windscale and Chernobyl accidents.

In simple models, an average value of deposition velocity is taken to be 10^{-3} ms^{-1} for all airborne activity (except the noble gases which do not deposit). However, strictly speaking, this value is appropriate for a $1 \mu\text{m}$ diameter aerosol passing over grassland. It is well known that deposition rates in woodland areas can be as much as a factor of 100 greater, whereas deposition in urban areas (roads, pavements) can be a factor of 10 smaller. Thus, concentrations of deposited activity are likely to vary by several orders of magnitude over small distances due to local heterogeneities. It follows that simple models can only estimate average levels of deposited activity.

The results of studies into the Chernobyl accident confirm that a value of 10^{-3} ms^{-1} is a reasonable average to use for aerosols on grass.

9.4.2.3 Wet Deposition

Wet deposition arises from rainout (whereby the pollutant is incorporated into raindrops during the formation of the droplet) and washout (whereby ready-formed raindrops take up the pollutant as they fall). Washout, in particular, depends on the particle size (for aerosols), the chemical form (gases and vapours) and the rainfall rate

Experiments suggest that the washout coefficients vary by several orders of magnitude for different airborne pollutants. Moreover, simple atmospheric models of wet deposition suffer from a number of limitations, the most significant being the assumption that rainfall intensity is uniform over the entire plume. In fact, rainstorms are highly structured and there are large variations in intensity leading to localised 'hot spots'. Thus, there are large uncertainties in estimating this effect.

Finally, run-off should also be mentioned. In urban areas, prolonged rainfall can remove up to 90% of deposited activity into drains, sewers, etc. One could argue that, following a release of radioactivity, the 'best' weather conditions would be dry weather during the passage of the plume followed by heavy rain to wash away any deposited surface contamination.

9.5 Exposure Pathways from Gaseous Discharges

There are various ways in which individuals may accrue a radiation dose from airborne and deposited activity as listed in Table 9.1 and shown in Figure 9.2

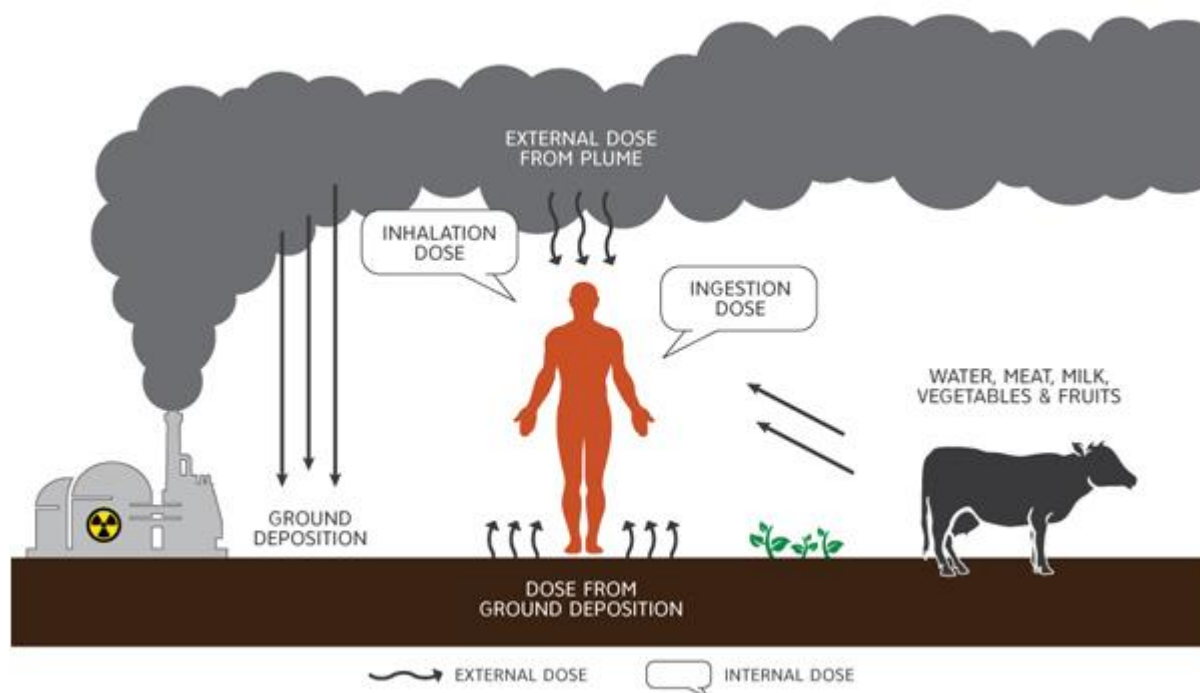


Figure 9.2 - Exposure pathways

Table 9.1 - Exposure Pathways

Pathway	Description
Cloudshine	External exposure to gamma and beta radiation from the cloud
Inhalation	Internal exposure from the inhalation of radioactivity
Groundshine	External exposure from to radioactivity deposited on the ground
Resuspension	Internal exposure from the inhalation of re-suspended activity
Personal contamination	External exposure from radioactivity which deposits on clothes or skin
Ingestion	Internal exposure from the ingestion of contaminated food / water

9.5.1 Cloudshine Dose

The dose accrued from exposure to gamma rays in a passing cloud can be estimated using the semi-infinite approximation. This assumes the individual to be immersed in a semi-infinite plume of uniform concentration. This gives a reasonable estimate of gamma dose from cloudshine at distances downwind where the cloud is well dispersed and the relative change in concentration is small over the spatial range of a gamma ray.

9.5.2 Inhalation Dose

The inhalation dose will depend on the time integrated air concentration (TIAC), the breathing rate and the relevant dose coefficient as shown in Equation 9.3.

$$\text{Dose (Sv)} = \text{TIAC}((\text{Bqsm}^{-3}) \times \text{Breathing rate (m}^3\text{s}^{-1}) \times \text{Dose Coefficient (SvBq}^{-1}) \quad 9.3$$

9.5.3 Groundshine Dose

The dose rate from gamma emitting material on the ground at any time depends on:

- the level and spatial distribution of contamination
- the energy of the gamma rays.

The dose accrued over a given exposure period will depend on:

- the half-life of the nuclide
- the rate of various removal processes (e.g. migration into soil).

In principle, the calculation of gamma dose involves a two-dimensional integral over the contaminated surface, taking account of the variation in concentration. However, in many cases, it is reasonable to assume that the activity is distributed uniformly over an infinite flat plane. The dose is then given by Equation 9.4

$$\text{Dose (Sv)} = \text{Ground Contamination (Bqm}^{-2}\text{)} \times \text{Ground integrated dose factor (Sv/Bqm}^{-2}\text{)} \quad 9.4$$

9.5.4 Resuspension

The TIAC can be determined by applying an appropriate resuspension factor. The dose can then be calculated by using Equation 9.3.

9.5.5 Foodstuffs

In many cases, the uptake of radionuclides from the diet has been identified as the main route of entry of fission and activation products into man. Many physical and chemical processes govern the transport and behaviour of radionuclides in soil, plants and animals and it is therefore necessary to use compartment modelling of these transfer processes to predict the concentrations of radioactive materials in foodstuffs.

9.6 Dispersion of Liquid Discharges in the Marine Environment

As for airborne releases, dispersal models can be used to model the behaviour of radioactive material in the marine environment. These must account for the transport of the activity by tides, currents, winds and marine life. The greater numbers of variables involved in modelling aquatic releases is reflected in the bigger uncertainties in the modelling results.

9.7 Dose Assessment

9.7.1 Prospective Dose Assessment

Dispersion models are used by both the operator and the regulator to determine the environmental impact (doses both to human and non-human species) of gaseous and liquid radioactive discharges. The discharge limits are used as the source terms. This work will generally be carried out as part of the Permitting / Authorisation process but will also support BAT etc. studies.

9.7.2 Retrospective Dose Assessment

Environmental monitoring (see Section 6.9) results are used as the basis for modelling the impact of actual discharges into the environment. The dose to the **Representative Person**⁷ is determined and compared with the public dose limit of 1 mSv per year and the dose constraint from a single source of 0.3 mSv per year.

9.8 Environmental Monitoring

All operators conduct a routine programme of environmental monitoring. Independent monitoring is also carried out by the regulator (EA / SEPA) and/or the Food Standards Agency (FSA). This monitoring includes gamma dose rates, air particulate monitoring, sediment sampling and the sampling of vegetation and foodstuffs. The results of the monitoring surveys are published by the Operator and also annually by the FSA as a 'Radioactivity in Food and the Environment' Report https://www.foodstandards.gov.scot/downloads/RIFE_29_%282023%29.pdf.

9.9 Summary

Regulatory Framework - EPR16, regulated by the EA (England & Wales), EASR18 (Scotland) regulated by SEPA

BPM (Scotland) equivalent to **BAT** (England and Wales)

Exposure Pathways: cloudshine, inhalation, personal contamination, ground shine, resuspension, ingestion

Atmospheric Modelling - use Gaussian Plume Model (but limited), dry and wet deposition

Marine Modelling - needs to account for tides, currents, winds and marine life. Many variables result in very large uncertainties

Prospective and retrospective dose assessments completed based on use of models and environmental monitoring results, respectively.

⁷ An individual, either hypothetical or specific, whose dose can be used for determining compliance with the relevant dose constraint

References

ICRP Publication 60 '1990 Recommendations of the International Commission on Radiological Protection'

ICRP Publication 103 'The 2007 Recommendations of the International Commission on Radiological Protection'

Martin, Harbison, Beach and Cole 'An Introduction to Radiation Protection'

Johnson 'An Introduction to Health Physics'