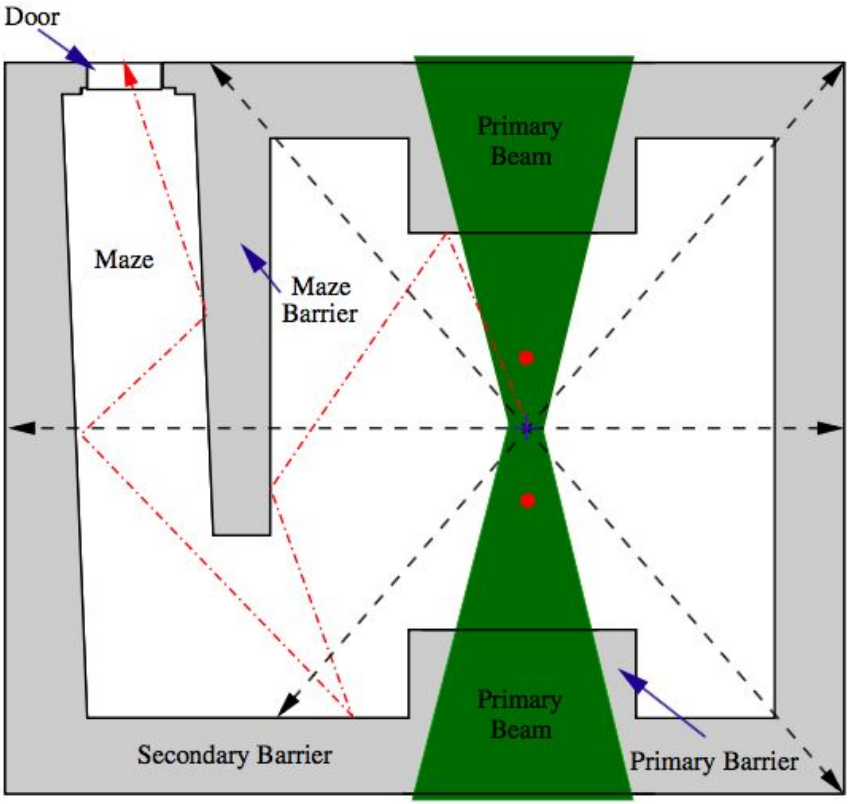
Health Physics for Medical Physicists



John Kildea
(Class Notes MDPH 316)
McGill University

Health Physics Notes - MDPH 613

Student Notes

John Kildea, PhD

Medical Physics Unit

McGill University

Montreal,Quebec

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CHAPTER 1
Introduction

1.1 Overview of Course

Health physics is the area of public health that concerns itself with radiation protection. In this course we will review the physics that governs the production and interaction of ionizing radiation and the biolological consequences of its interaction with living cells and tissue. We will then apply our knowledge to the safe use of ionizing radiation, as codified in national regulations and national and international memoranda. Although the concepts and principles of radiation protection that we will cover in this course relate to the use of ionizing radiation generally, we will focus in particular on the safe use of ionizing radiation in medicine.

1.2 Radiation and Radiation Protection

The use of ionizing radiation[1] offers great benefits to society. As detailed in table 1.1, these benefits are realized in medicine, science/academia, industry and power generation. However, radiation may be detrimental to human health and its use necessarily involves management of risk so as to mitigate injury to the individual user and to society in general. Radiation risk may be controlled but not eliminated; any use of radiation entails an associated non-zero risk. The goal of radiation protection is, thus, to minimize the risk while maintaining the overall benefit to society.

[1] From this point forward we will use the word radiation to mean ionizing radiation. This course does not deal with non-ionizing radiation.

Table 1.1: Examples of the beneficial uses of radiation in modern society.

Field	Example Applications
Medicine	Diagnostic radiology, nuclear medicine, radiation therapy, cardiology, interventional medicine, blood irradiation, sterilization
Science/academia	Teaching, research, plant and animal physiology, environmental studies, archaeology, oceanography, geology
Industry	Food sterilization, security services, reflective signs (tritium coated), smoke detectors, defect detection radiography, well-logging, spacecraft propulsion, pest control, waste management
Nuclear power	Electricity generation using nuclear fission. Accounts for ∼15% of electricity supply in Canada, ∼20% in the US and ∼80% in France

Table 1.2: Examples of public health concerns in society.

Category	Examples
Regulation	Smoking laws, drug laws, pollutant limits, radiation limits
Infectious diseases	Vaccination programs, awareness campaigns
Sanitation	Waste and sewage disposal, drinking water
Health promotion	Physical exercise, diet and nutrition, smoking, blood supply, breast cancer screening (mammography)

1.3 Health Physics

1.3.1 Public and Private Health

Public health is concerned with maintaining the health of the population. Private health, or clinical medicine, on the otherhand, aims to cure sick individuals. Whereas clinical medicine is the remit of individual clinicians, the responsibility for public health falls to society as a whole. Public health policy aims to prevent disease and ill health through community efforts, regulations and health promotion. Table 1.2 lists some examples of situations that concern public health policy.

1.3.2 Health Physics and Medical Physics

Radiation plays an important role in both public health and clinical medicine. Health physics is the discipline of public health that concerns itself with radiation protection. Medical physics is the clinical use of radiation to diagnose and cure disease in private health. As shown in figure 1.1, health physics encompasses aspects of radiation physics, radiobiology, public health policy and emergency response preparedness.

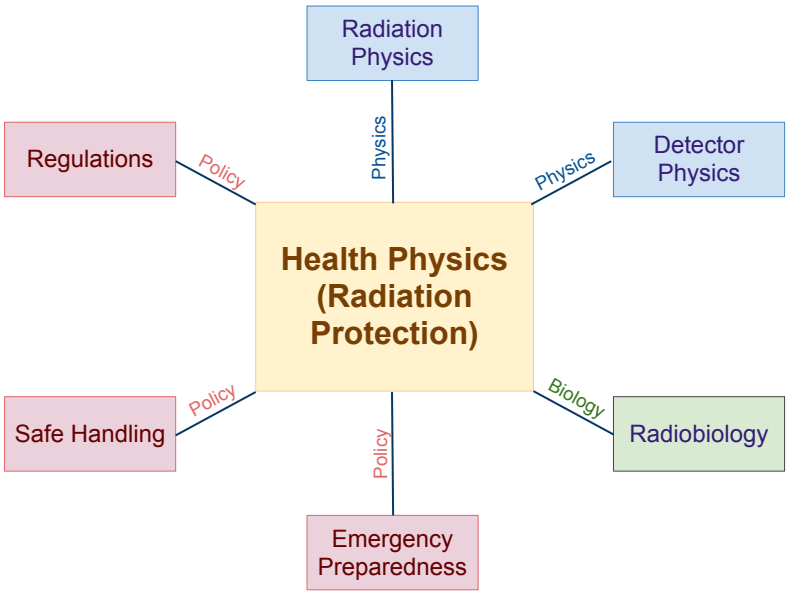


Figure 1.1: An overview of the subjects encompassed by health physics.

1.3.3 Health Physicists

Health physicists are scientists trained in radiation physics who apply the principles of radiation protection to situations where individuals and populations may be exposed to radiation (both ionizing and non-ionizing). In order to perform his/her duties a health physicist should:

- Understand what radiation is and how it interacts with matter/tissue
- Know how to detect and quantify radiation
- Understand the risks and the biological effects of radiation exposure
- Know the regulations pertaining to radiation protection
- Know how to safely handle radioactive sources
- Implement safe practises so as to minimize the risks of radiation exposure in accordance with the ALARA (As Low As Reasonably Achievable) principle

The goal of this course in Health Physics is to provide an introduction to each of the points listed above. Figure 1.2 provides a conceptual overview of the material that is covered in this course.

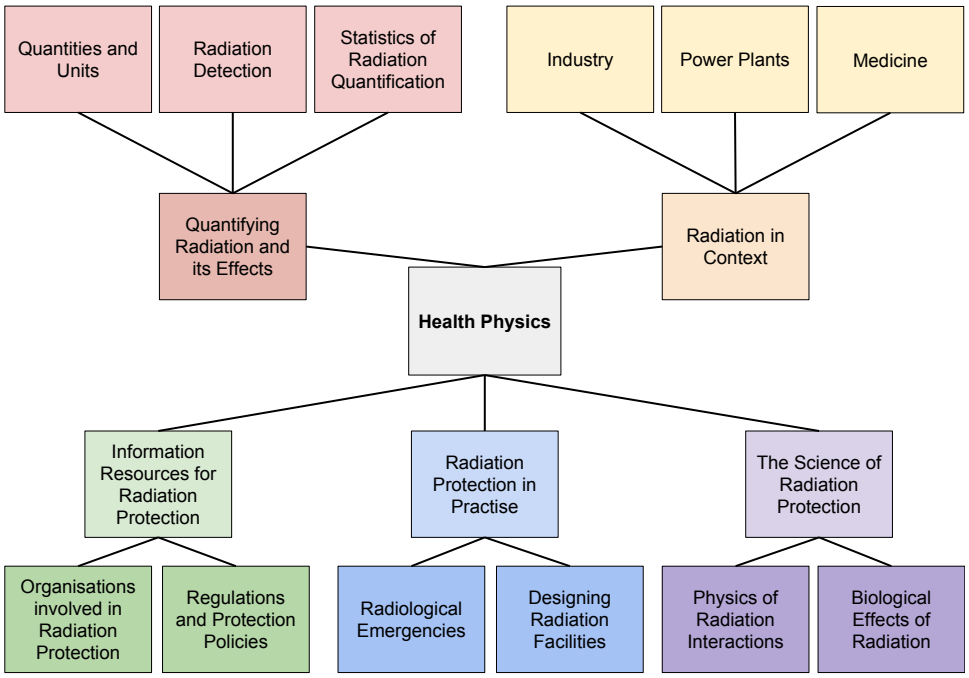


Figure 1.2: A concept map of the Health Physics material covered in this course.

CHAPTER 2
Radiation, its Origins and Interactions

The goal of this chapter is to review the physics of radiation and its interactions. We will discuss what radiation is, how it is produced, how we classify it and how it interacts with matter. We will also examine the sources of radiation that society is exposed to. By the end of the chapter we should be able to describe radiation to a member of the public and use our understanding of radiation to explain the physics behind the three tenets of radiation protection—distance, time and shielding.

2.1 Definition of Radiation

The term radiation has its origin in the Latin word radius, which was the name for the spoke of a wheel. Radii (or rays) pointed outward from the center of the wheel. Likewise, the term radius describes the distance from the center of a circle to its circumference. Radiation, in the modern sense, describes a stream of particles that emanate outward from a source. The particles may be elementary or composite and can be charged or uncharged. However, all radiating particles share the property that, as they move away from their source, they carry kinetic energy that was imparted to them at the source.

The concept of radiation radiating out from a source was incorporated into the original ionizing radiation warning sign. Known as the trefoil sign, it was first used at the University of California Radiation Laboratory in Berkeley in 1946. In 2007, the International Atomic Energy Agency (IAEA) drew up a new ionizing radiation warning sign that incorporates the original but which is supposed to better signify the danger of radiation (as opposed to resembling

(a) The original sign. (b) The new IAEA sign.

Figure 2.1: The old and new ionizing radiation warning signs.

a benign propeller!). The old and new ionizing radiation warning signs are shown in figure 6.1.

2.2 The Inverse Square Law

The inverse-square law is a simple but important practical manifestation of the definition of radiation. Using the definition of radiation as a stream of particles carrying energy that emanate outwards from a source, we can understand the inverse-square law.

Take for example two geocentric spheres α and β, as shown in figure 2.2. β has a larger radius than α ($r_\beta > r_\alpha$). If a point source of radiation is placed at the centre of the spheres, then the radiation fluence ϕ (defined as number of radiation particles per unit area N/A) measured on the surface of α will be higher than that measured on the surface of β ($\phi_\alpha > \phi_\beta$). The reasoning is straightforward. Because the radiation is moving outwards equally in all directions it will have spread out more by the time it reaches the surface of β than it had when it reached the surface of α. Since the surface area of a sphere is $4\pi r^2$ and since the same number of particles N must cross both surfaces, we can show that

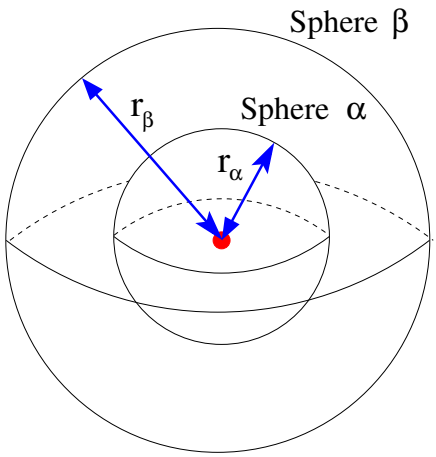


Figure 2.2: Illustration of the physical basis for the inverse-square law. If point source of radiation is held at the center of the two concentric spheres α and β, then the radiation particle fluence measured on the surface of β will be less than that measured on the surface of α due to the spreading out (dilution) in all directions of the radiation as it "radiates" outwards from the source.

$$\begin{aligned} N &= \phi_\alpha A_\alpha = \phi_\beta A_\beta \\ \Rightarrow \phi_\alpha 4\pi r_\alpha^2 &= \phi_\beta 4\pi r_\beta^2 \\ \Rightarrow \frac{\phi_\beta}{\phi_\alpha} &= \frac{r_\alpha^2}{r_\beta^2} \end{aligned} \tag{2.1}$$

If, for example, $r_\beta = 2r_\alpha$, we can see that $\phi_\beta = \phi_\alpha/4$. This holds for any sectional area projected outward from the source onto the two surfaces.

The inverse-square law is important when dealing with radiation. It is particularly important in health physics as it represents a straightforward way in which one's radiation exposure can be significantly reduced—simply by standing back. The inverse-square law strictly only holds for point sources of radiation but it is generally true in practice unless one is in close proximity to a large source.

2.3 Ionization and Ionizing Radiation

2.3.1 The Bohr-Rutherford Atomic Model

An atom is the basic unit of matter. In the Rutherford-Bohr atomic model, an atom comprises a central, dense, positively charged nucleus containing nucleons (at least one positively charged proton and zero or more neutral neutrons) surrounded by a cloud of negatively charged electrons that are bound to the atom by the attractive positive charge of the nucleus. The electron cloud is arranged into discrete shells, with electrons in the inner shells being more tightly bound to the nucleus than those in the outer shells.

2.3.2 Ionization

Under normal circumstances, the positive charge of the nucleus and the negative charge of the electrons balance one another making the atom neutral. However, if sufficient kinetic energy is given to a bound electron, it may overcome the attractive electromagnetic force of the nucleus and escape the atom, leaving behind a shell vacancy. The process of electron escape is known as ionization and an ionized atom is called an ion. The energy required to eject an electron from an atom is known as the electron's binding energy. Electronic binding energies are quantized according to atomic shell. The ionization potential of an atom refers to the energy of its least bound electron, i.e., the minimum energy needed to eject an electron from the atom. Ionization potentials range from a few electron volts for the alkali elements to 24.6 eV for helium.

2.3.3 Ionizing Radiation

Radiating neutral or charged particles with sufficient energy to eject electrons from the matter they encounter are called ionizing radiation. The process of

Table 2.1: The eight ways by which an atom may be ionized.

Method of Ionization	Source of ionizing radiation
Photoelectric effect	Photon external to atom
Compton scattering	Photon external to atom
Triplet production	Photon external to atom
Auger effect	Internal electron rearrangement
Electron capture	Internal nucleus/electron rearrangement
Internal conversion	Internal nucleus/electron rearrangement
Coulomb interaction	External particle
Positron annihilation	External particle

ionization is either direct or indirect. Direct ionization results from the interaction of charged particles with matter (also called Coulomb interactions). Indirect ionization happens when a neutral particle interacts with matter to produce a charged particle that in turn causes ionization. Radiation carries energy and through the process of ionization, the energy of the radiating particles is deposited in the matter through which they pass.

There are eight methods through which ionization (i.e., the production of an atomic shell vacancy) may occur, as shown in table 2.1. Three of them (photoelectric effect, Compton scattering, and triplet production) involve photons impinging on the atom, three others are the result of internal rearrangements amongst the atomic electrons or within the nucleus (Auger effect, electron capture, internal conversion) and two arise from charged particles incident on the atom (Coulomb interaction, positron annihilation). All are of interest to the health physicist.

2.4 Classification of Ionizing Radiation

In addition to being classified as charged (directly ionizing) or uncharged (indirectly ionizing), ionizing radiation may also be categorized by place of origin and by particle type and energy. Figure 2.3 shows the physical classification scheme for radiation.

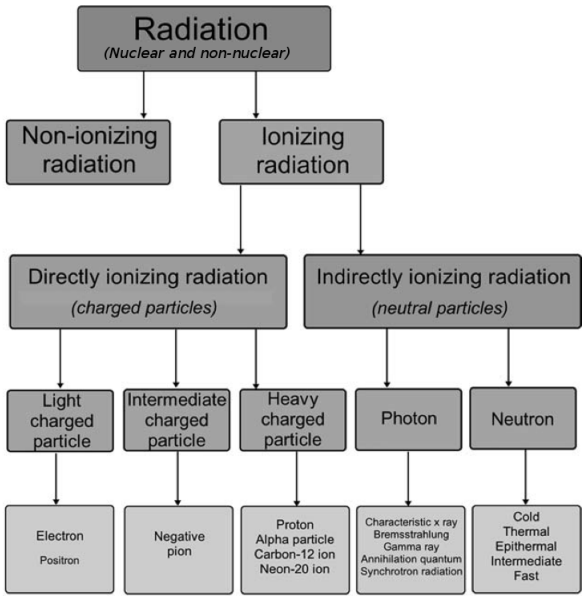


Figure 2.3: Classification of radiation. Figure adapted from Podgoršak (2010).

2.4.1 Radiation by Place of Origin

In terms of origin, nuclear radiation arises from within the nucleus and all other radiations arise from outside the nucleus.

Nuclear Radiation

Nuclear radiation includes alpha, beta and gamma rays plus internal conversion electrons.[1] Alpha particles are helium nuclei (2 protons, 2 neutrons), beta particles are electrons or positrons and gamma rays are photons. Alpha particles and conversion electrons are emitted with discrete energies, whereas beta particles share their energies with neutrinos, thereby exhibiting a spectrum of possible emission energies. Figure 2.4 presents the discrete spectral lines for a number of alpha-emitting radioactive sources and 2.5 shows the

[1] Spontaneous fission neutrons should also be listed under nuclear radiation but since spontaneous fission is a very unlikely event and only occurs for heavy elements with $Z^2/A \geq 15$, we won't concern ourselves with them here.

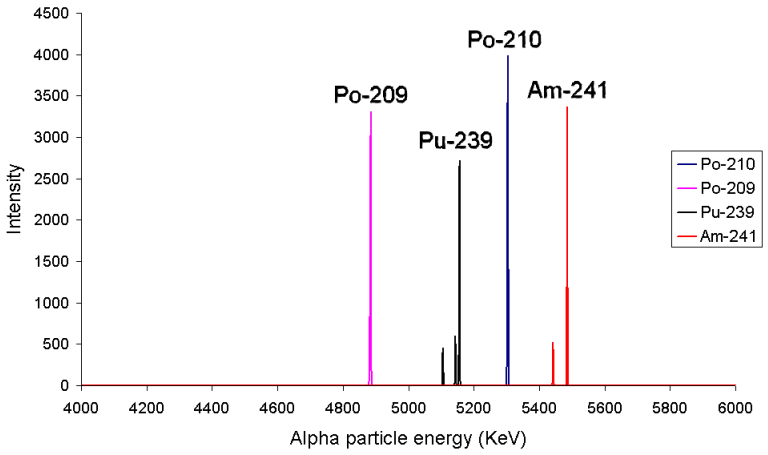


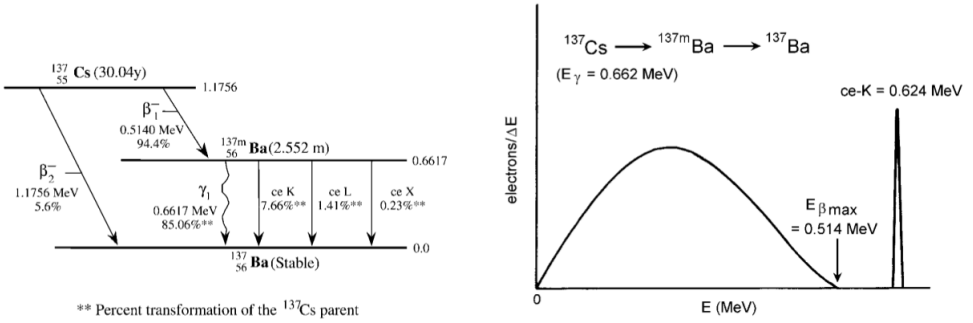
Figure 2.4: Spectral lines of four alpha-emitting radionuclides. Note that, for some radionuclides, the daughter nucleus may be left in an excited state such that the alpha particle does not receive the maximum energy available. The remaining energy is typically emitted almost immediately ($<12^{-12}$ s) as a gamma ray. Figure from wikipedia.org.

nuclear decay scheme (gamma, beta and conversion electrons) and associated electron energy spectrum of Caesium-137.

Non-nuclear Radiation

Ionizing photons produced outside of the nucleus are referred to as X rays, to distinguish them from gamma rays, which arise within the nucleus[2] . Beams of non-nuclear particulate radiation are typically labelled according to the particles that comprise them, for example proton beams or carbon-ion beams.

[2] Historically, the term gamma ray described photons with energies higher than diagnostic X rays. However, with the advent of high-energy linear accelerators that energy distinction became blurred. Today, we recognize photons from the nucleus as gamma rays and photons from beyond the nucleus as X rays. In astrophysics, photons of extraterrestrial origin with energies above a few hundred keV are still referred to as gamma rays, regardless of their (often unknown) origin.



(a) Nuclear decay scheme.

(b) Electron spectrum.

Figure 2.5: The nuclear decay scheme and electron energy spectrum of Cs-137. The continuous beta particle spectrum and the discrete line resulting from K-shell conversion-electron are clearly seen. Figures from Martin (2006)

2.4.2 Indirectly Ionizing Radiation

Indirectly ionizing radiation comprises neutral particles; either photons or neutrons.

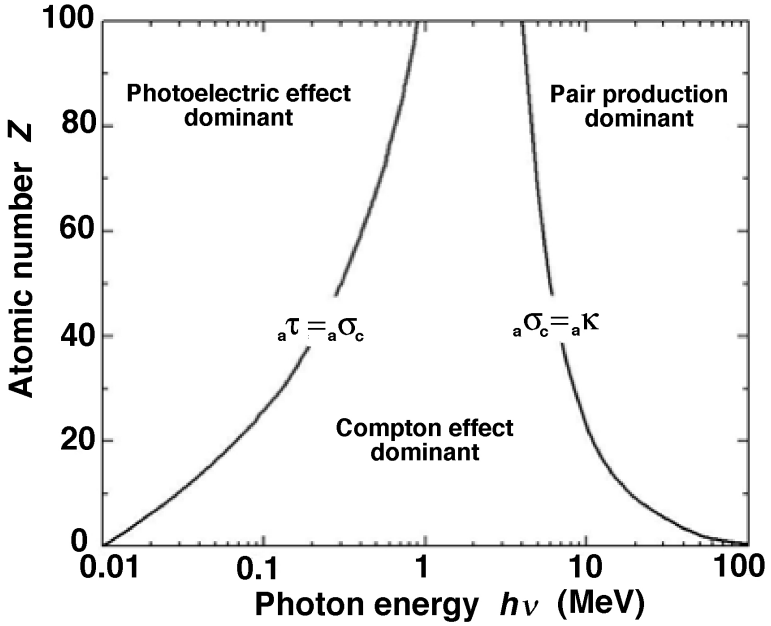
Photons

Photons have no mass and are electromagnetic waves. They may be categorized according to their energy in the electromagnetic spectrum.

Photons indirectly ionize matter by means of the photoelectric effect, Compton scattering or pair production. The probability of a particular interaction occurring depends on (a) the energy of the photon $h\nu$, and (b) the atomic number Z of the absorber material. Figure 2.6 shows the regions of predominance of the three interactions as a function of $h\nu$ and Z.

With regard to photon interactions of interest to the health physicist, photodisintegration, although not a means of ionization, is also an important interaction. Photodisintegration (also known as the photonuclear interaction or nuclear photoeffect) is energetically feasible above ∼10 MeV. In the photodisintegration interaction a photon is absorbed by a nucleus and the most likely outcome is release of a single photoneutron, through the (γ, n) reaction.

Figure 2.6: Regions of relative photon interaction predominance as a function of $h\nu$ and Z. Figure from Podgoršak (2010)



Release of other charged particles, gamma rays, fission products or additional neutrons is also possible but much less likely.

Neutrons

Neutrons have mass ($\sim 2000\mathrm{m_e}$) and may be sub-categorized by kinetic energy into eight groups:

1. Ultracold ($E_\mathrm{K} < 2 \times 10^{-7}$ eV)
2. Very cold (2×10^{-7} eV $\leq E_\mathrm{K} \leq 5 \times 10^{-5}$ eV)
3. Cold (5×10^{-5} eV $\leq E_\mathrm{K} \leq 0.025$ eV)
4. Thermal ($E_\mathrm{K} \simeq 0.025$ eV)
5. Epithermal (1 eV $\leq E_\mathrm{K} \leq 1$ keV)
6. Intermediate (1 keV $\leq E_\mathrm{K} \leq 0.1$ MeV), and
7. Fast (0.1 MeV $< E_\mathrm{K} <$ few MeV)
8. High-energy neutrons ($E_\mathrm{K} >$ few MeV)

Neutrons are indirectly ionizing particles that interact through nuclear interactions. Dose deposition by neutron beams is a two step process in which (a) the neutrons produce charged particles by nuclear interactions and (b) the charged particles deposit dose by Coulomb interactions.

Thermal, epithermal and fast neutrons have applications in medicine. Thermal neutrons are used in boron-neutron capture therapy (BNCT) and fast neutrons are used in external beam radiation therapy. Fast neutrons are also used for *in-vivo* neutron activation analysis (for example, to quantify the amount of calcium in the body for the diagnosis of osteopenia or osteoporosis) and neutron radiography.

In industry, cold, thermal and hot neutrons are used in scattering and radiography experiments to study the properties and the structure of materials. Applications are found in the academic sciences (condensed matter physics, biology, solid state chemistry, geology, mineralogy, etc) and in the nuclear, aerospace and weapons industries. Neutrons are also produced and used in nuclear power reactors.

High-energy neutrons have no direct application but are encountered in the atmosphere, resulting from cosmic ray interactions, and at high-energy particle accelerators.

There are five main processes through which neutrons may interact with matter. They are elastic (neutron deflected without energy loss) and inelastic (neutron deflected with change in energy) scattering, neutron capture, spallation and fission. The first three have important application in the moderation (slowing down) and absorption of neutrons within the shielding of high-energy radiation therapy installations. Neutron production and shielding in radiation therapy will be discussed in detail later in this course.

2.4.3 Directly Ionizing Radiation

Numerous charged particles fall under the category of directly ionizing radiation. In general three main groups are recognized, according to mass: light, intermediate and heavy. Electrons/positrons are light charged particles. Pions and muons are considered to have intermediate mass and all charged particles with mass equal to or greater than the proton are considered heavy.

2.4.4 Light Charged Particles

Electrons and other charged particles interact through Coulomb interactions. Coulomb interactions are either collisional or radiative. In collision interactions (also known as ionization interactions), the charged particle energy is lost to the absorbing medium, whereas in radiation interactions it is lost to bremsstrahlung photons. Collision losses occur when the incident charged particle interacts with orbital electrons. Collisions can be hard (impact parameter of the order of the atomic radius) or soft (impact parameter much greater than the atomic radius). Hard collisions result in an ionized target atom, whereas soft collisions cause atomic excitation. Radiation loss occurs when the incident charged particle is decelerated by the Coulomb field of the target nucleus. Hard, soft and radiation collisions are illustrated in Figure 2.7.

In medicine, electron beams are used in external beam radiation therapy for the treatment of superficial lesions. In society, electron beams are most frequently encountered in cathode ray tubes (CRTs), and until the advent of solid-state electronics, in vacuum tubes. Although the electron beams generated by the CRTs of TVs and computer monitors are not in themselves a radiation hazard, the bremsstrahlung beams ($\sim$40 kV) they produce may be. Indeed, the US Food and Drug Administration regulates that the X-ray radiation from TVs should not exceed 0.5 mR/hr. Modern TVs and computer monitors employ liquid crystal displays and so are not a radiation hazard.

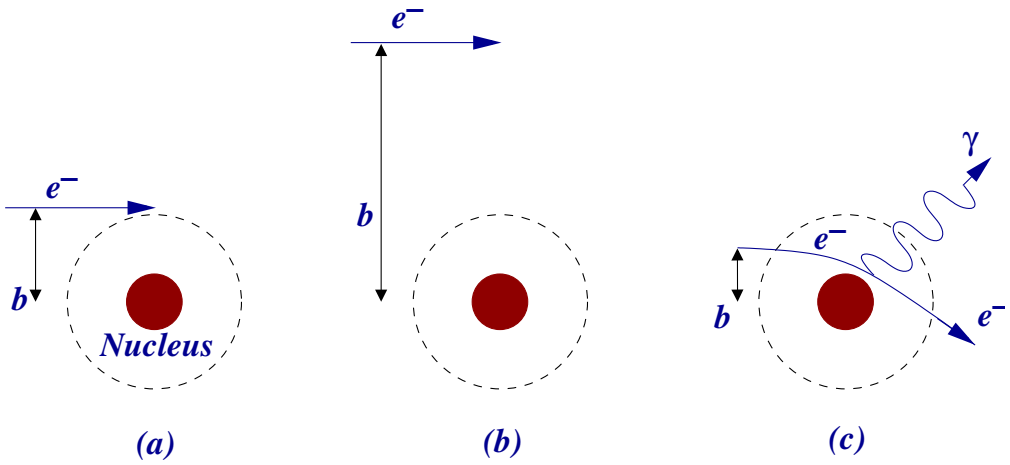


Figure 2.7: Collision types in the Coulomb interaction: (a) Hard collision, in which the impact parameter b is of the order of the atomic radius, (b) Soft collision, in which $b \gg a$, and (c) Radiation collision, where $b \ll a$.

2.4.5 Intermediate Mass Charged Particles

Muons are elementary charged particles (leptons) with a mass of $207m_e$ and a mean lifetime of 2.197 μs. They are found naturally in cosmic rays and may be generated artificially by particle accelerators.

Pions are intermediate mass charged particles (mesons) that can be charged or neutral. Charged pions have a mass of $273\mathrm{m}_e$ and neutral pions have a mass of $264\mathrm{m}_e$. Like muons, charged and neutral pions are produced naturally in cosmic ray air showers. However, unlike muons, their very short mean lifetimes (2.6×10^{-8} s for charged pions and 0.83×10^{-16} s for neutral pions), mean that they cannot survive to sea level. Negative pions produced at high-energy accelerator laboratories (for example, at the TRIUMF cyclotron in British Columbia) have found limited application in radiation therapy.

The cosmic-ray muons that constantly bombard the Earth's surface ($\sim$10 000 $m^{-2} \cdot s^{-1}$) form a significant portion of our background radiation exposure. Muons are produced as secondary particles in the air showers (atmospheric charged particle cascades) that result when high-energy primary cosmic rays

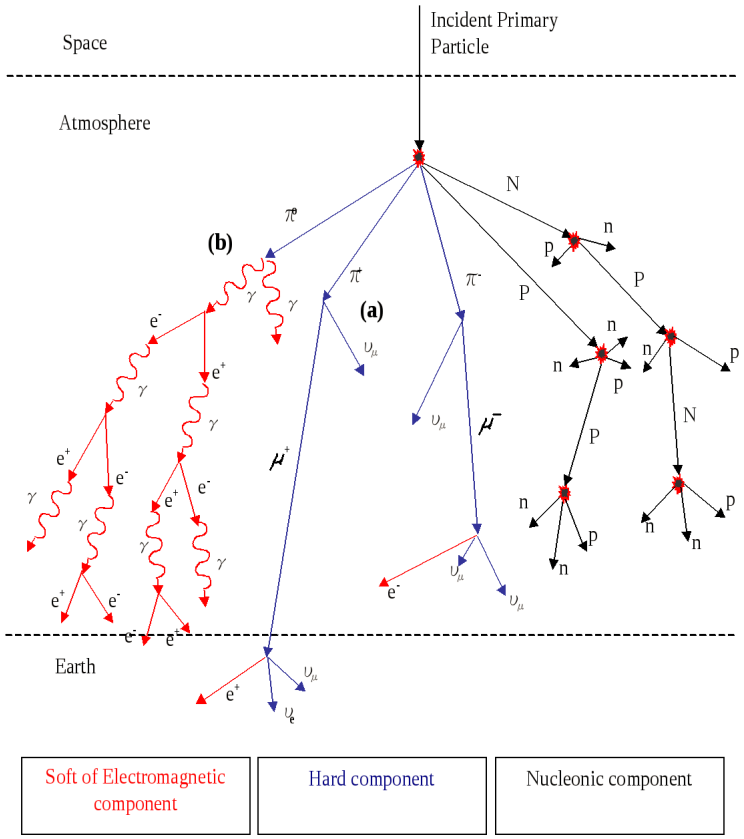


Figure 2.8: Schematic of a cosmic ray air shower, showing the soft (electromagnetic), hard (penetrating) and nucleonic components. Cosmic ray muons form a significant portion of the background radiation at the Earth's surface.

(typically protons) interact with air in the upper atmosphere. Figure 2.8 shows how an incident primary cosmic ray forms an air shower.

2.4.6 Density of Ionization

In addition to classification based upon the physical properties of its radiating particles, an ionizing beam may be classified in terms of the density of ionization it produces on passing through a medium. Beams may be more or less densely ionizing, depending on the particle type and energy involved. The term Linear Energy Transfer (LET) is used in health physics and radiobiology to specify the density of radiation produced in an absorber material by a beam of ionizing radiation passing through it. In general, as shown table 2.3 and

Table 2.2: LET values for various high and low LET radiation beams. Table from Podgoršak (2010).

Low LET radiation	LET (keV/μm)	High LET radiation	LET (keV/μm)
X rays: 250 kVp	2	Electrons: 1 keV	12.3
γ rays: Co − 60	0.3	Neutrons: 14 MeV	12
X rays: 3 MeV	0.3	Protons: 2 MeV	17
Electrons: 10 keV	2.3	Carbon ions: 100 MeV	160
Electrons: 1 MeV	0.25	Heavy ions:	100–2000

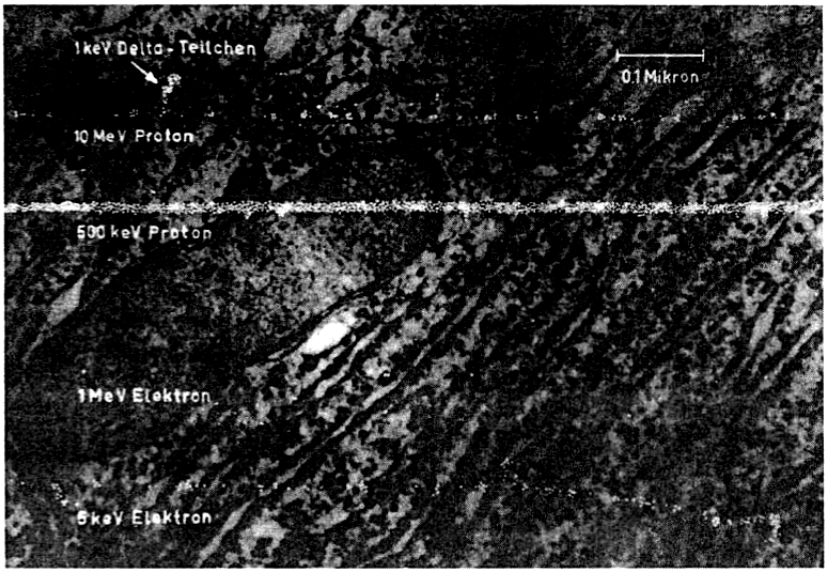


Figure 2.9: Ionization density associated with different types of radiation. The background is an electron micrograph of a human cell. The white dots are a computer simulation of ionization tracks. Figure from Hall and Giaccia (2006).

illustrated in figure 4.6, heavier and lower-energy particles have greater LET values than lighter and higher-energy particles.

Two categories of LET are recognized:

- Low LET (sparsely ionizing) radiation (below 10 keV/μm)
- High LET (densely ionizing) radiation (above 10 keV/μm)

LET is discussed further in chapter 3.

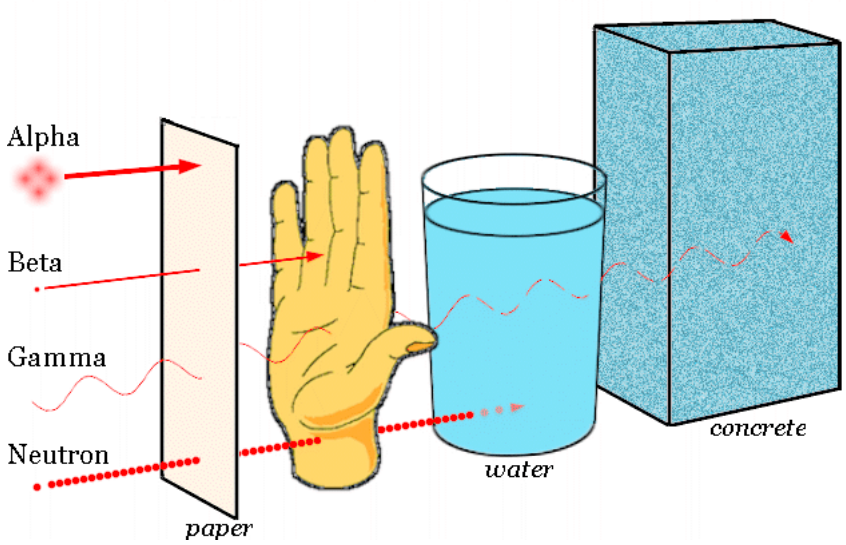


Figure 2.10: Illustration of the penetrability of various radiation beams. Figure from Baylor College of Medicine website.

Radiation Penetrability

Particle type and energy also govern how deeply a radiation beam may penetrate into an absorber material. An energetic beam of light particles will take longer to loose all of its initial energy and so may penetrate deeper. In general, alpha particles (higher LET) penetrate the least and gamma rays (lower LET) penetrate the deepest, as shown in figure 2.10. Neutron beams are moderated (ie loose energy) on passing through low-Z materials. Water, for example, is a good attenuation material for neutrons due to its hydrogen content.

The concepts of Half Value Layer (HVL), Tenth Value Layer (TVL), linear attenuation coefficient (μ) and range (R) are used to describe the penetrability of a radiation beam. They will be discussed in more detail in chapter 3.

2.5 Background Radiation Exposure

In radiation protection, it is instructive to consider the level of background radiation, both natural and artificial, that society is exposed to. It must always be accounted for when measuring the level of radiation from a source and it provides a good indicator of what is "normal" when discussing radiation

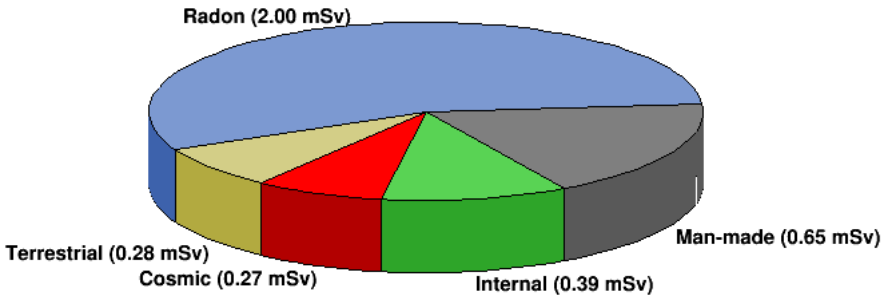


Figure 2.11: Breakdown of the estimated annual background radiation for a member of the population of the United States. Note: the corresponding effective dose values for Canada are estimated at approximately 20% lower. Data from NCRP Report 93 (1987).

exposure. Figure 2.11 shows the distribution of the estimated background radiation for a member of the population of the United States (NCRP Report 93, 1987; NCRP Report 94, 1987). The total background effective dose is estimated at 3.6 mSv per year (the corresponding effective dose values for Canada are estimated to be approximately 20% lower).

2.5.1 Natural Background Radiation

Radiation is a natural part of the Earth's environment and so it is ubiquitous and unavoidable. Natural background radiation is composed of cosmic radiation (from space), terrestrial radiation (from the soil and the rocks) and internal radiation (from the radioactive elements within our bodies). Terrestrial and internal radiations are either cosmogenic or primordial in origin. Our exposure to natural radiation is a function of our location on the Earth's surface—the abundance of radioactive elements varies over the Earth's surface and the cosmic ray flux increases with altitude and with distance from the equator.

Cosmic Radiation

Cosmic radiation comprises high-energy particles and ionizing photons from the Sun and the Galaxy. The vast majority of cosmic ray particles that enter the Earth's atmosphere (ie primary cosmic rays) interact in the upper atmosphere and do not make it to sea level. However, as illustrated by figure 2.8 above, the interaction by-products (ie secondary cosmic rays) may be penetrating enough to reach the surface.

At sea level, cosmic ray muons make up about 10% of the natural radiation background. At higher altitudes the cosmic ray flux is higher, such that cosmic rays represent an important radiological hazard for air crew and astronauts. Indeed, in Europe, airline crew are classified as radiation workers. Figure 2.12 shows how the radiation dose due to cosmic rays increases as a function of altitude. Owing to the presence of the Earth's magnetic field and to the fact that primary cosmic rays are mainly positively charged protons, the cosmic ray flux also varies as a function of latitude on the Earth's surface—the lowest flux is found at equatorial levels with the highest flux found near the magnetic poles.

Cosmogenic Radioactivity

Cosmogenic radioactivity is radioactivity that is naturally and continuously produced in the atmosphere by the interactions of primary cosmic rays with air. Many cosmogenic radionuclides are produced. However, only two, tritium and carbon-14 are of significance in health physics.

Tritium (a beta emitter with a half-life of 12 years) readily forms tritiated water, which is chemically indistinguishable from normal water but is hazardous when ingested. It is important to understand the source of natural tritium since tritium is widely used commercially in products that incorporate self-powered lighting (wrist watches, exit signs, etc).

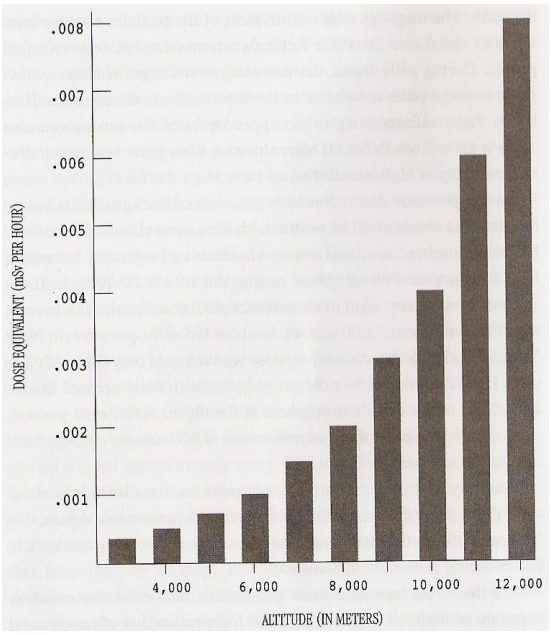


Figure 2.12: The increase in cosmic radiation exposure as a function of altitude above sea level. These data are for a latitude of 60° north. Figure from Barish (2009).

Atmospheric thermal neutrons, produced in cosmic ray interactions, may interact with nitrogen-14 to produce the radionuclide carbon-14. Carbon-14 decays through beta emission with a half-life of 5 730 years. It is used in radiocarbon dating of carbonaceous materials up to about 60 000 years old. Carbon dating is based upon the premise of continuous production of carbon-14 in the atmosphere by cosmic rays. While this is believed to be true in archaeological terms, nuclear bomb tests in the atmosphere between 1955 and 1980 dramatically changed the amount of recent carbon-14 such that significant corrections are required in order to date organic material formed since 1950.

Primordial Radioactivity

Primordial radioactivity is radioactivity that has been around since the formation of the Earth (about 4.5 billion years ago). It is believed that all the heavy elements were created in supernova explosions and that it was their differing

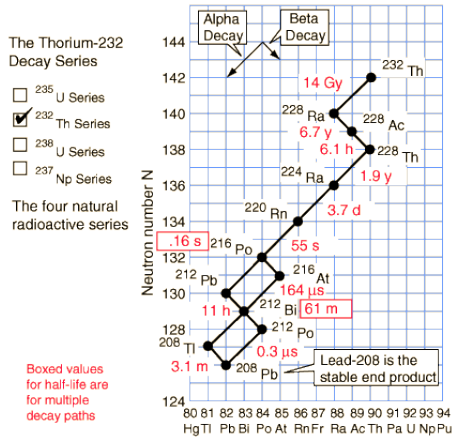
half-lives that has resulted in their present-day abundances. Most sources of primordial radioactivity belong to three of four radioactive series, as shown in figure 2.13. The series are:

- The Thorium series - originating with thorium-232 ($T_{\frac{1}{2}}$=14 $\times$ 10^9 years) and ending in lead-208
- The Actinium series - originating with uranium-235 ($T_{\frac{1}{2}}$=0.7 $\times$ 10^9 years) and ending in lead-207
- The Neptunium series - originating with neptunium-237 ($T_{\frac{1}{2}}$=2 $\times$ 10^6 years) and ending in bismuth-209
- The Uranium series - originating with uranium-238 ($T_{\frac{1}{2}}$=4 $\times$ 10^9 years) and ending in lead-206

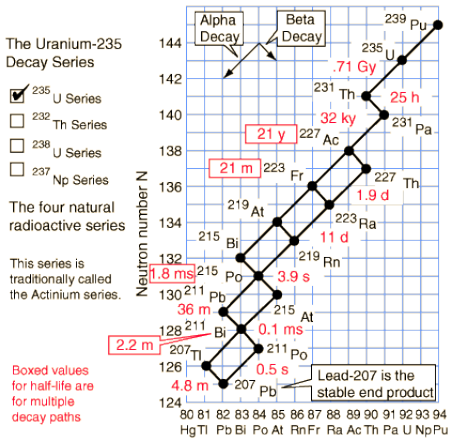
The series are characterized by an unstable parent nucleus with a very long half life and by a final stable lead-isotope nucleus, for the thorium, actinium and uranium series, and the stable bismuth-209 nuclide for the neptunium series. Due to the relatively short half-life of neptunium-237, the neptunium series is no longer found in nature, having decayed away since the formation of the Earth.

In addition to the three remaining high-Z radioactive series, several low-Z primordial radionuclides are found in nature. These include, for example, potassium-40, rubidium-87, and lanthanum-138, among others.

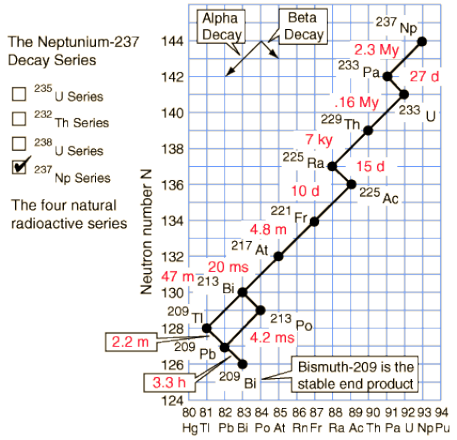
The primordial radionuclides are important in health physics as they form the largest contribution to the background radiation at sea level. They are found in the soil (for example, uranium) in the air (for example, radon, which is formed in all three radioactive series) and in our bodies and foods (for example, potassium-40). In fact, potassium-40 is the largest source of internal radiation and bananas, potatoes and other fruit/vegetables containing potassium often



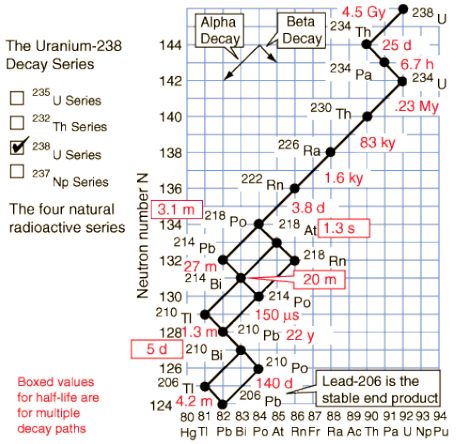
(a) The Thorium series.



(b) The Actinium series.



(c) The Neptunium series.



(d) The Uranium series.

Figure 2.13: The four radioactive series. Figures from the HyperPhysics website.

set off security radiation detectors at ports and border crossings. As shown in figure 2.11, radon alone is believed to account, on average, for half of the background radiation (both natural and man-made) at sea-level.

Enhanced Natural Background Radiation

Enhanced sources of natural background radiation are sources that are natural in origin but which result in increased levels of exposure due to human activity. Examples include enhancement of cosmic radiation by flying at high-altitude and enhancement of radon by living in insulated buildings where radon gas levels may build up.

2.5.2 Artificial Background Radiation

In addition to natural background radiation, modern society is exposed to radiation from artificial (man-made) sources. These include radiation exposure for medical purposes, low-level radiation exposure from nuclear power plants and radiation exposure from consumer goods (such as smoke detectors, wrist-watches and CRT TV screens). Unlike natural background radiation, which cannot be controlled, artificial background radiation levels can be controlled and so are generally regulated by radiological or environmental authorities.

Anthropogenic Background Radiation

Artificial background radiation that does not include medical exposure (which can be considered private exposure) is referred to as anthropogenic background radiation. Many national and international bodies seek to understand, measure and reduce the levels of anthropogenic background radiation. These bodies will be discussed in chapter 6.

Medical Background Radiation

Radiation exposure for medical purposes accounts for the vast majority of modern society's exposure to man-made radiation.

2.6 The Physics of Radiation Protection

2.6.1 Distance, Time and Shielding

The three tenets underlying the physics of radiation protection are: distance, time, and shielding. Each is described below:

(1) Distance. The distance from a radiation source should be maximized. The inverse-square law governs the fall-off in dose as a function of source distance; meaning that, for example, a doubling of the distance will reduce the exposure level by a factor of 4.

(2) Time. The duration of an exposure should be minimized, since the accumulated exposure increases linearly as a function of time.

(3) Shielding. The amount of shielding around a radiation source should be maximized and optimized. Radiation beams loose energy (by ionizations and excitations) and so are attenuated on passing through material. The level of attenuation depends on the type and energy of the radiation beam and the atomic number and density of the absorbing material.

2.6.2 Scatter Radiation

As described in section 2.3.3, radiation may be absorbed or scattered when it interacts with matter. The interaction process depends on the type and energy of the radiation and on the absorbing material. Scattering of radiation by air is generally not a major concern but radiation scattered of the floor, the source container, the walls and the furniture may be considerable. Scattered radiation must always be considered when dealing with radiation. Radiation scattered back into a beam, which would otherwise not be detected, tends to increase a measured radiation signal. In general, the level of radiation at a point is a function of **distance, time, shielding and scatter**.

2.6.3 Bremsstrahlung

An energetic beam of radiating charged particles such as beta rays or electrons may be completely stopped in an absorber material but may nevertheless represent a radiation risk due to secondary bremsstrahlung radiation. For example, the bremsstrahlung X rays produced by energetic beta particles are generally much more penetrating than the beta particles that produced them. Since bremsstrahlung beams are more efficiently produced in high-Z materials, beta-emitting radionuclides should be doubly-shielded, first by an inner layer of low-Z material to absorb the beta particles, followed by an outer layer of high-Z material to absorb the secondary X rays.

A similar double-shield is required to guard against gamma rays produced by neutron capture events in the door of a high-energy radiation therapy room. Neutron doors will be discussed later in the course.

CHAPTER 3
Quantification and Detection of Radiation

In order to manage the exposure of individuals and society to radiation, radiation levels must be measurable and their physical and biological effects quantifiable. Radiation dosimetry is the science of measuring and quantifying radiation exposure in a way that allows for a meaningful dose-based assessment of its biological effects. Radiobiology is the science of understanding those biological effects.

In this chapter we will examine both the quantification and the detection of radiation. By the end of the chapter we will be able to quantify radiation both from a general physics point of view and from a radiation protection perspective. We will be able to list the various types of radiation detector available and, if presented with a particular radiation exposure scenario, determine the most suitable radiation detector to use in order to quantify the exposure.

3.1 Radiation Quantification

The "strength" of a particular radiation may quantified from a number of different perspectives. Physicists and engineers working with radiation for physical purposes (for example at a nuclear power plant or in a radiation laboratory) are interested in purely physical quantities, such as the amount of the radioactive substance or the rate at which it decays. Medical physicists are generally interested in the radiation dose deposited in living tissue or in a phantom. Health physicists are similarly interested in dose to tissue but they are concerned not just with single tissues or individuals but rather with

multiple tissues, populations and perhaps multiple radiations and multiple exposures. The choice of radiation quantity to use in any particular situation depends on the type of radiation involved and the purpose of the measurement.

3.1.1 Physical Quantities

Physical quantities used in radiation physics allow for quantification of the strength of a radiation source (or radiation field) without reference to biological systems. As such, they are free of biological uncertainties and are generally straightforward to measure. Table 3.1 lists the physical quantities of interest in radiation physics. For convenience, a distinction is made between quantities concerning radiation generally and quantities concerning radioactivity.

Fluence and Flux

Particle fluence Φ is defined as the number of particles (charged particles or photons) crossing unit area

$$\Phi = \frac{\mathrm{d}N}{\mathrm{d}A}$$

Particle flux, or fluence rate $\dot{\Phi}$ is defined as the number of particles crossing unit area per unit time.

$$\dot{\Phi} = \frac{\mathrm{d}\Phi}{\mathrm{d}t} = \frac{\mathrm{d}}{\mathrm{d}t}\left(\frac{\mathrm{d}N}{\mathrm{d}A}\right)$$

Particle fluence has units of m^{-2} and particle flux has units of $\mathrm{m}^{-2} \cdot \mathrm{s}^{-1}$

For a monoenergetic beam, the energy fluence Ψ is the product of the particle fluence and the energy of the particles, and the energy flux $\dot{\Psi}$ (also known as the intensity) is the energy fluence per unit time. The unit of energy fluence is $\mathrm{J} \cdot \mathrm{m}^{-2}$ and the unit of energy flux is $\mathrm{J} \cdot \mathrm{m}^{-2} \cdot \mathrm{s}^{-1}$

For polyenergetic beams, the particle fluence and energy fluence spectra are respectively defined as

Table 3.1: Physical quantities used to quantify radiation.

Quantity	Definition	Formula	Unit
	Concerning Radiation Beams		
Particle fluence	Number of particles crossing per unit area for a monoenergetic beam	$\Phi = \frac{\mathrm{d}N}{\mathrm{d}A}$	m^{-2}
Particle flux	Number of particles crossing per unit area per unit time for a monoenergetic beam	$\dot{\Phi} = \frac{d\Phi}{dt}$	$\mathrm{m}^{-2} \cdot \mathrm{s}^{-2}$
Energy fluence	Product of particle fluence and particle energy for a monoenergetic beam	$\Psi = \frac{dN}{dA}E = \Phi E$	$\mathrm{J} \cdot \mathrm{m}^{-2}$
Energy flux	Product of particle flux and particle energy for a monoenergetic beam	$\dot{\Psi} = \frac{d\Psi}{dt}$	$\mathrm{J} \cdot \mathrm{m}^{-2} \cdot \mathrm{s}^{-1}$
Particle fluence spectrum	Particle fluence as a function of energy for a polyenergetic beam	$\Phi_E(E) = \frac{d\Phi}{dE}(E)$	
Energy fluence spectrum	Energy fluence as a function of energy for a polyenergetic beam	$\Psi_E(E) = \frac{d\Phi}{dE}(E)E$	
Exposure	Amount of charge of either sign collected in a given mass of air at standard temperature and pressure (for photons less than 3 MeV only)	$X = \frac{\Delta Q}{\Delta m_{\mathrm{air}}}$	$\mathrm{C} \cdot \mathrm{kg}_{\mathrm{air}}^{-1}$
Kerma	Mean energy transferred from indirectly to directly ionizing radiation per unit mass of absorbing material	$K = \frac{\mathrm{d}E_{tr}}{\mathrm{d}m}$	$\mathrm{J} \cdot \mathrm{kg}^{-1}$
Linear attenuation coefficient	The probability of a photon beam interacting in an absorber material as a function of depth into the material	$\mu = \frac{ln2}{\mathrm{HVL}}$	cm^{-1}
Mass attenuation coefficient	The linear attenuation coefficient divided by the density of the absorber material	$\mu_m = \frac{\mu}{\rho}$	$\mathrm{cm}^2 \cdot \mathrm{g}^{-1}$
Half Value Layer	The depth into an absorber material at which the intensity of a radiation beam drops to half of its initial value	$\mathrm{HVL} = \frac{ln2}{\mu}$	cm
Tenth Value Layer	The depth into an absorber material at which the intensity of a radiation beam drops to one tenth of its initial value	$\mathrm{TVL} = \frac{ln10}{\mu}$	cm
Linear Energy Transfer	The energy absorbed per unit length by an absorbing medium as ionizing radiation moves through it	$\mathrm{LET} = \frac{\mathrm{d}E}{\mathrm{d}l}$	$\mathrm{keV} \cdot \mu\mathrm{m}^{-1}$
Linear stopping power	The energy lost by a charged particle (or beam of charged particles) per unit length as it (they) traverses an absorber material	$\mathrm{S} = -\frac{dE}{dx}$	$\mathrm{MeV} \cdot \mathrm{cm}^{-1}$
Mass stopping power	The linear stopping power divided by the density of the absorbing medium	$\frac{S}{\rho}$	$\mathrm{MeV} \cdot \mathrm{cm}^2 \cdot \mathrm{g}^{-1}$
	Concerning Radioactive Sources		
Activity	Number of radioactive transformations per second	$\mathcal{A} = \frac{\mathrm{d}N}{\mathrm{d}t}$	Becquerel (1 Bq = 1 s^{-1})
		$\mathcal{A}(t) = \mathcal{A}(0)e^{-\lambda t}$	Curie (1 Ci = 3.7×10^{10} Bq) Rutherford (1 Rd = 1×10^6 Bq)
Radioactive decay constant	The probability of a radioactive decay transformation per unit time	$\lambda = \frac{ln2}{t_{1/2}}$	s^{-1}
Half life	Time necessary for half the original number of nuclei in a radioactive sample to decay	$t_{1/2} = \frac{ln2}{\lambda}$	s
Specific activity	The activity of a radioactive sample divided by its mass	$a = \frac{\mathcal{A}}{m}$	$\mathrm{Ci} \cdot \mathrm{g}^{-1}$ (SI unit: $\mathrm{Bq} \cdot \mathrm{kg}^{-1}$)
Exposure rate constant	The exposure rate, in R/h, at a distance of 1 m from a sample of a radionuclide having an activity of 1 Ci	Γ	$\mathrm{R} \cdot \mathrm{m}^2 \cdot \mathrm{Ci}^{-1} \cdot \mathrm{h}^{-1}$
Exposure rate due to a radionuclide	The exposure rate constant for the radionuclide times its activity, divided by the distance from the source squared	$\dot{X} = \frac{X}{t} = \Gamma\frac{\mathcal{A}}{d^2}$	$\mathrm{R} \cdot \mathrm{h}^{-1}$

$$\text{Particle fluence spectrum} = \Phi_E(E) = \frac{\mathrm{d}\Phi}{\mathrm{d}E}(E)$$

$$\text{Energy fluence spectrum} = \Psi_E(E) = \frac{\mathrm{d}\Phi}{\mathrm{d}E}(E)E$$

Integration of the area under the particle fluence spectrum and under the energy fluence spectrum yields the total particle fluence and total energy fluence, respectively, of a polyenergetic beam.

Exposure

Exposure X is a radiometric (as opposed to dosimetric) unit that allows for quantification of the fluence of an x-ray or gamma-ray field. It is defined as the amount of charge *of either sign* collected in a given mass of air at standard temperature and pressure (STP). Owing to the difficulty of collecting all the charge from the large masses of air required at very high photon energies, exposure is defined only for photons with energy less than 3 MeV.

$$X = \frac{\Delta Q}{\Delta m}$$

The unit of exposure is coulombs per kilogram of air. The old unit of exposure was the röntgen, which was defined as the amount of radiation required to liberate charge of either sign amounting to one electrostatic unit of charge (esu) in 1 cm^3 of dry air at STP. As shown in the box below, 1 R works out at 2.58×10^{-4} $\mathrm{C} \cdot \mathrm{kg}_{\mathrm{air}}^{-1}$.

Given that 1 esu is 3.335×10^{-10} C and given that the density of air at STP is 1.293×10^{-3} $\mathrm{g/cm}^3$, we can see that

$$X(1\ \mathrm{R}) = \frac{\Delta Q}{\Delta m} = \frac{3.335 \times 10^{-10}\ \mathrm{C}}{1.293 \times 10^{-6}\ \mathrm{kg}} = 2.58 \times 10^{-4} C/kg$$

where 1.293×10^{-6} kg is the mass of 1 cm^3 of air at STP.

Kerma

Kerma stands for Kinetic Energy Released per unit MAss. It quantifies the mean energy transferred, per unit mass of absorbing material, from indirectly ionizing radiation to directly ionizing radiation. Kerma does not concern itself with whether or not the directly ionizing radiation that is produced is ultimately absorbed in the absorbing material. For example, the secondary electrons produced subsequent to a Compton interaction may be fully absorbed by the material, may partially leave the material or may loose energy to bremsstrahlung photons that may leave the material. KERMA only considers the initial transfer of energy from the incident photon to the Compton electron.

$$K = \frac{\mathrm{d}E_{tr}}{\mathrm{d}m}$$

Kerma is of interest in medical physics when dealing with photons and neutrons. Photons transfer energy to electrons and fast neutrons transfer energy to scattered nuclei. The unit of KERMA is the gray, corresponding to 1 $\mathrm{J{\cdot}kg^{-1}}$.

Exposure and Kerma are related through the quantity called air-KERMA K_{air}, as derived in the box below.

Given the definitions of exposure X and Kerma K, we can relate the two quantities in air.

$$X = \frac{\Delta Q}{\Delta m_{\text{air}}} \Rightarrow \Delta m_{\text{air}} = \frac{\Delta Q}{X}$$

and

$$K_{\text{air}} = \frac{\Delta E_{\text{tr}}}{\Delta m_{\text{air}}} \Rightarrow \Delta m_{\text{air}} = \frac{\Delta E_{\text{air}}}{K_{\text{air}}}$$

so

$$\frac{\Delta Q}{X} = \frac{\Delta E_{\text{tr}}}{K_{\text{air}}} \Rightarrow K_{\text{air}} = \frac{\Delta E_{\text{tr}}}{\Delta Q} X$$

where K_{air} is in units of Gy and X is in units of R.

We now ask, what is $\frac{\Delta E_{\text{tr}}}{\Delta Q}$ the energy transferred in producing an ion-pair in air?

We can use the quantity $\bar{W}_{\text{air}}$, which is the average energy required to produce an ion pair in air, having a currently-accepted value of 33.97 eV per ion-pair.

Since, the definition of exposure specifies the charge of either sign (ie one or the other) produced in air, we can divide $\bar{W}_{\text{air}}$ by e/ion-pair, where e is the charge of the electron in coulombs.

Thus

$$\frac{\bar{W}_{\text{air}}}{e} = 33.97 \frac{eV \cdot ion - pair}{e \cdot ion - pair} = 33.97 \frac{J}{C}$$

We now have an expression for $\frac{\Delta E_{\text{tr}}}{\Delta Q}$, which we can use in the expression for K_{air}

$$K_{\text{air}} = 33.97 \frac{J}{C} X$$

If X has a value of 1 R ($= 2.58 \times 10^{-4} \frac{C}{kg}$) then

$$K_{\text{air}} = (33.97 \frac{J}{C})(2.58 \times 10^{-4} \frac{C}{kg}) = 0.00876 \frac{J}{kg} \simeq 0.01\ Gy$$

So, an exposure of 1 R is equivalent to Air Kerma of approximately 0.01 Gy.

The Linear Attenuation Coefficient

The linear attenuation coefficient μ is described as the probability per unit path length that a photon will have an interaction with the material through which it is passing. It is synonymous with the radiation decay constant λ and may be derived in a similar manner (see below). The linear attenuation coefficient has units of cm^{-1}. Its value depends on the energy of the photon and on the density *rho* and atomic number Z of the absorbing material.

The Beer-Lambert law governs the exponential attenuation of a photon beam as it passes through an absorbing material. In terms of μ, the Beer-Lambert law may be expressed as

$$I(x) = I(0)e^{-\mu x}, \tag{3.1}$$

where $I(x)$ is the intensity of the photon beam at depth x in the material and $I(0)$ is the incident intensity.

The linear attenuation coefficient is normally determined using narrow-beam geometry, in which the beam under study arises from a narrowly collimated source. Although narrow beams are seldom encountered in applied radiation physics, they are practically useful, since they are more reproducible than broad beams.

A linear attenuation coefficient is defined for all photon interaction types, with the overall coefficient μ equalling the sum of the individual coefficients:

$$\mu = \sigma_{\mathrm{R}} + \tau + \sigma_{\mathrm{C}} + \kappa_{\mathrm{p}} + \kappa_{\mathrm{t}} + \sigma_{\mathrm{PN}} \tag{3.2}$$

where

- σ_{R} is the attenuation coefficient for Rayleigh scattering
- τ is the attenuation coefficient for the photoelectric effect
- σ_{C} is the attenuation coefficient for Compton scattering

- κ_{p} is the attenuation coefficient for pair production
- κ_{t} is the attenuation coefficient for triplet production
- σ_{PN} is the attenuation coefficient for photodisintegration

The mass attenuation coefficient μ_m is equal to the linear attenuation coefficient divided by the mass per unit volume (ie density ρ) of absorber material. Since the mass attenuation coefficient is independent of the density of the absorber material it is a useful quantity when comparing the Z dependencies of photon attenuation in various absorber materials. The SI unit for μ_m is m^2/kg, although the older unit of cm^2/g is more often used in practice.

Half Value Layer and Tenth Value Layer

The Half Value Layer (HVL) is the depth into an absorber material at which the intensity of a radiation beam drops to half of its initial intensity. It is synonymous with the half-life of a radioactive material (see below). Likewise, the Tenth Value Layer (TVL) is the depth at which the incident radiation level is reduced to one tenth of its initial value. The relationship between the HVL and the TVL is derived in the box below.

As is the case for half-life and the radioactive decay constant, the relationship between HVL and TVL and the linear attenuation coefficient is straightforward to derive. The end results are

$$HVL = \frac{ln2}{\mu} \tag{3.3}$$

$$TVL = \frac{ln10}{\mu} \tag{3.4}$$

Since $HVL = \frac{ln2}{\mu}$ and $TVL = \frac{ln10}{\mu}$,

$$\mu = \frac{ln2}{HVL} = \frac{ln10}{TVL}$$

$$\Rightarrow \frac{HVL}{TVL} = \frac{ln2}{ln10}$$

Both HVL and TVL are frequently encountered in health physics. They are most often used in the context of shielding design.

Multiple HVLs (HVL_1, HVL_2, HVL_3, etc) are defined. For a monoenergetic beam, each HVL is of equal depth. For a polyenergetic beam, however, beam hardening (or softening) will result in longer (or shorter) consecutive HVLs. Similarly for TVL.

In beam hardening, lower energy photons are preferentially removed from the beam as it is attenuated, leaving more penetrating higher-energy photons behind. In beam softening, it is the higher energy photons that are removed. Beam softening is seldom an issue in health physics but it may be encountered when using very high-energy photon beams and high-Z materials, for which pair-production may preferentially remove higher-energy photons. Figure 3.1 below shows how the mass attenuation coefficient μ/ρ for lead varies as a function of photon energy. At low energies, μ/ρ decreases with increasing energy but at high energies (above about 2 MeV) it increases slowly as a function of energy.

Linear Energy Transfer

Linear energy transfer (LET) is defined as the rate of *energy absorption* by an absorbing medium as ionizing radiation moves through the medium.

$$\text{LET} = \frac{\text{d}E}{\text{dl}}$$

LET has units of keV/μm. It was discussed qualitatively in section **??**.

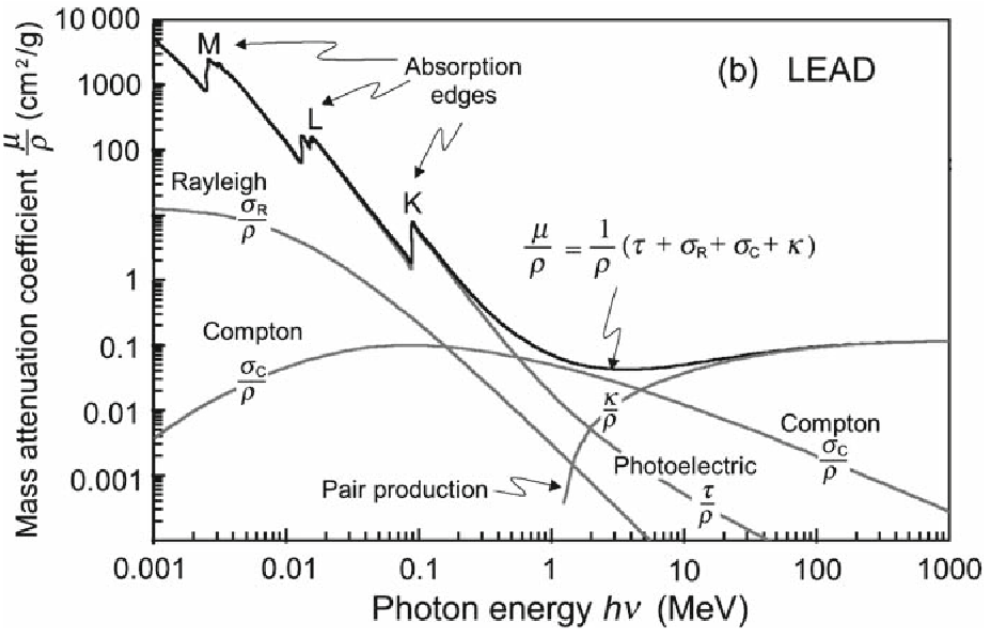


Figure 3.1: The mass attenuation coefficient for lead as a function of photon beam energy. Photon beams with energy spectra in the pair-production regime may undergo beam softening. Figure from Podgoršak (2010).

Stopping Power

In contrast to LET, which focuses on the absorbing medium, stopping power S focuses on the energy lost by the radiation. The stopping power of a charged particle (or beam of charged particles) is the energy lost by the particle (or beam) per unit length as it traverses an absorbing material.

$$\mathrm{S} = -\frac{\mathrm{d}E}{\mathrm{dx}}$$

Stopping power may be greater than LET as the secondary electrons produced in the medium as the charged particle moves through it may have enough energy to leave the region of interest and deposit their energy elsewhere. Thus, the energy absorbed by the medium (LET) in the region of interest will be less than the energy lost by the charged particle (stopping power). Stopping power is normally expressed in units of MeV/cm.

The mass stopping power $\frac{S}{\rho}$ is equal to the (linear) stopping divided by the density of the absorbing medium. The units of $\frac{S}{\rho}$ are $\frac{\text{MeV}\cdot\text{cm}^2}{\text{g}}$ (ie $\frac{\text{MeV/cm}}{\text{g/cm}^3}$).

Activity

The *activity* $\mathcal{A}$ of a radionuclide is defined as the number of radioactive transformations (decays) it undergoes per second.

$$\mathcal{A} = -\frac{\mathrm{d}N}{\mathrm{dt}}$$

The SI unit for activity is the becquerel (Bq), named after Henri Becquerel who discovered radioactivity in 1896 and for which he shared the Nobel prize in physics with Pierre and Marie Curie in 1903.

$$1 \ \text{Bq} = 1\frac{\text{decay}}{\text{second}}$$

In radiation physics, the curie (Ci) is often used as an alternative unit of radioactivity. It was named in honour of Pierre Curie at the 1910 Radiology congress. The curie is much larger than the becquerel and originally one curie indicated the activity of one gram of radium-226. Today it is simply defined as

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

which, using modern measurement techniques, equates to the activity of 0.998 g of radium-226.

The rutherford (Rd) is a unit of activity defined as 1 million transformations per second (1 Rd = 1×10^6 Bq). It was named after Ernest Rutherford and provides a unit of activity between the bequerel and the curie.

The Law of Radioactive Decay

The activity of a radioactive substance decreases as a function of time—it becomes iteratively weaker as the decay process itself depletes the number

of unstable nuclei available for future decay. As such, one can state that the (decreasing) rate of change in the number of nuclei (ie the activity) is proportional to the number of nuclei present

$$\mathcal{A} = -\frac{\mathrm{d}N}{\mathrm{d}t} \propto N$$

$$\Rightarrow \mathcal{A} = -\frac{\mathrm{d}N}{\mathrm{d}t} = \lambda N$$

where $\mathcal{A}$ is the activity, $-\mathrm{d}N/\mathrm{d}t$ is the rate of decrease in the number N of radioactive nuclei at any time t and λ is a constant of proportionality known as the *radioactive decay constant*. Rearranging an integrating the above equation we get

$$\int_{N_0}^{N(t)} \frac{\mathrm{d}N}{N} = -\lambda \int_0^t \mathrm{d}t,$$

which, when integrated gives

$$lnN(t) - lnN(0) = -\lambda t$$

this may be further rearranged to give

$$lnN(t) = -\lambda t + lnN(0)$$

which, as shown in figure 3.2, is recognized as the equation of a straight line with slope $-\lambda$ and intercept $lnN(0)$.

Using the rules of logarithms, we can see that

$$ln\left(\frac{N(t)}{N(0)}\right) = -\lambda t$$

which, using the definition of a logarithm, gives

$$\frac{N(t)}{N(0)} = e^{-\lambda t}$$

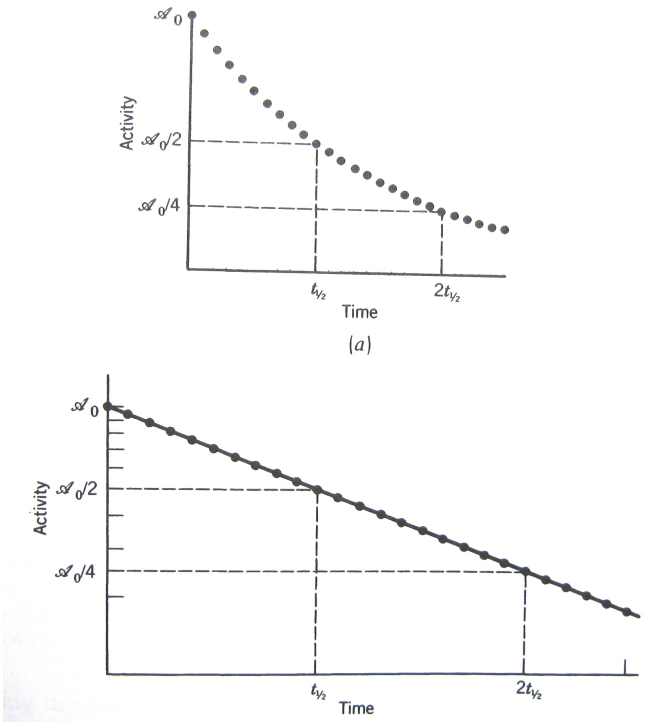


Figure 3.2: Radioactive decay. The activity of a radioactive sample decreases exponentially as a function of time.

multiplying both sides by λ and recognizing that $\mathcal{A} = \lambda N$, we obtain the law of radioactive decay

$$\mathcal{A}(t) = \mathcal{A}(0)e^{-\lambda t} \tag{3.5}$$

Put simply, the law of radioactive decay says that the activity $\mathcal{A}(t)$ of a radioactive sample at any time $t = t$ is equal to the initial activity of the sample $\mathcal{A}(0)$ at time $t = 0$ multiplied by an exponential decay factor $e^{-\lambda t}$, where λ is the decay constant of the particular radionuclide. Since the exponential decay factor is unitless, λ has units of s^{-1}. It gives the probability of radioactive decay per unit time.

Half-Life

The half-life $t_{\frac{1}{2}}$ of a radionuclide is defined as the time necessary for half of the original nuclei in a sample of it to decay. Since we are more accustomed to thinking in terms of time, rather than inverse time, half-life is a more

meaningful unit than the decay constant, although they both essentially tell us the same thing. The relationship between half-life and decay constant can be easily derived from the definition for half-life, as shown in the box below.

At time $t = t_{\frac{1}{2}}$, $\mathcal{A} = \frac{\mathcal{A}(\prime)}{2}$,

$$\text{thus} \quad \frac{\mathcal{A}}{\mathcal{A}(0)} = \frac{1}{2} = e^{-\lambda t_{\frac{1}{2}}}$$

which, taking the logarithm of both sides, gives

$$ln1 - ln2 = -\lambda t_{\frac{1}{2}}$$

$$\Rightarrow t_{\frac{1}{2}} = \frac{ln2}{\lambda} \tag{3.6}$$

Specific Activity

The specific activity a of a radioactive sample is defined as its activity per unit mass.

$$a = \frac{\mathcal{A}}{m}$$

where m is the mass of the sample. The units of specific activity are Ci/g.

Carrier-Free Specific Activity

Carrier-free specific activity (CFSA) refers to the specific activity of a pure sample of a radionuclide, ie a sample that is free of impure "carrier" material. In principle it is impossible to obtain a pure sample of a radionuclide due to continuous decay. However, theoretically one can calculate the CFSA for each radionuclide, as shown in the box below.

$$\text{CFSA} = \frac{\mathcal{A}}{m} = \frac{\lambda N}{m}$$

For one mole of pure radionuclide, we can say that $N = N_A$ and $m = \mathcal{M}$, where N_A is Avogadro's number and $\mathcal{M}$ is the atomic mass of the radionuclide.

$$\Rightarrow \text{CFSA} = \frac{\lambda N_A}{\mathcal{M}} \tag{3.7}$$

Exposure Rate Constant

The exposure rate constant Γ is defined as the exposure rate, in R/h, at a distance of 1 m from a sample of a radionuclide having an activity of 1 Ci. Its unit is $\frac{R \cdot m^2}{Ci \cdot h}$. The exposure rate constant accounts for photons only but includes all photons arising from decay of the nucleus, including gamma rays, characteristic X rays (following electron capture) and bremsstrahlung photons (following beta decay).

Each radionuclide has a unique value of Γ that may be calculated by considering all of the photon producing decay modes, their energies and their attenuations in air. Table 3.2 presents Γ values for some commonly used radionuclides. Knowing Γ one can determine the expected exposure rate of a source at a given distance from the source.

$$\dot{X} = \frac{X}{t} = \Gamma \frac{\mathcal{A}}{d^2}$$

Considering the definition of Γ, one can see that it incorporates the three tenets of radiation protection, distance, time and shielding (by air in this case) and the activity of a radioactive sample.

Air Kerma Rate Constant

The Air Kerma rate constant Γ_{AKR} is defined as the Air Kerma rate, in Gy/h, at a distance of 1 m from a sample of a radionuclide having an activity of

ISOTOPE	Γ	
	$\frac{\text{R-m}^2}{\text{Ci-h}}$	Air kerma $\left(\frac{\text{Sv-m}^2}{\text{MBq-h}}\right)$
Antimony-122	0.24	5.68E−08
Cesium-137	0.33	7.82E−08
Chromium-51	0.016	3.77E−09
Cobalt-60	1.32	3.13E−07
Gold-198	0.23	5.44E−08
Iodine-125	0.07	1.66E−08
Iodine-131	0.22	5.20E−08
Indium-192	0.48	1.14E−07
Iron-59	0.64	1.52E−07
Mercury-203	0.13	3.08E−08
Potassium-42	0.14	4.73E−08
Radium-226	0.825	1.96E−07
Sodium-22	1.20	2.84E−07
Sodium-24	1.84	4.35E−07
Zinc-65	0.27	6.39E−08

From *Radiological Health Handbook*. Rev ed. Rockville, MD: US Public Health Service, Bureau of Radiological Health; 1970.

Table 3.2: Exposure rate constants for a number of radionuclides. Figure from Cember and Johnson (2009).

1 MBq. In Health Physics its unit is $\frac{Sv \cdot m^2}{MBq \cdot h}$. It provides the same information as the exposure rate constant but expressed in more modern units. Like the exposure rate constant, it accounts for photons only but includes all photons arising from decay of the nucleus, including gamma rays, characteristic X rays (following electron capture) and bremsstrahlung photons (following beta decay). Table 3.2 presents Γ_{AKR} values for some commonly used radionuclides.

3.1.2 Dosimetric Quantities

In the early days of radiation usage, dosimetric quantities (quantities used to quantify dose to tissue) were poorly defined. Two very simple quantities were employed. The first was known as the *paper-clip unit* and it referred to the amount of radiation required to produce a detectable shadow of a paper clip on a piece of film. A radiation worker would carry a piece of film with a paper clip while carrying out his duties and examine it at the end of each day to quantify his exposure. For larger radiation doses, such as those used in therapy, the *skin erythema unit* was defined. It was the amount of radiation needed to produce skin redness. Nowadays we can recognize that neither quantity is really useful as neither is biologically meaningful and both are energy dependent.

Table 3.3: Dosimetric quantities and units used in radiation protection. These quantities apply equal to external and internal sources of exposure.

Quantity	Definition	Formula	Unit
	Purely Physical		
Absorbed dose	Energy absorbed per unit mass	$D = \frac{\Delta E_{abs}}{\Delta m}$	Gray (1 Gy = 1 $\frac{J}{kg}$)
	Concerning Individuals		
Equivalent dose	Sum of absorbed doses D to a single exposed tissue or organ T for one or more radiations R, with each D multiplied by the appropriate radiation weighting factor	$H_T = \sum_R w_R D_{T,R}$ (for radiations R)	Sievert (1 Sv = 1 $\frac{J}{kg}$)
Effective dose	Sum of equivalent doses H_T to one or more exposed tissues and organs T, with each H_T multiplied by the appropriate tissue weighting factor	$E = \sum_T w_T H_T$ (for tissues T)	Sievert
Personal dose equivalent	The equivalent dose in soft tissue below a specified point on the body at an appropriate depth d (0.07 mm or 10.0 mm below skin, 3.0 mm into eye)	$H_p(d)$	Sievert
	Concerning Populations		
Collective equivalent dose	Product of average equivalent dose to organ T and the number of individuals exposed	$S_T = \sum_i \bar{H}_{T,i} N_i$ (for individuals i)	Person-sievert
Collective effective dose	Product of average effective dose and the number of individuals exposed	$S = \sum_i \bar{E}_i N_i$ (for individuals i)	Person-sievert

In modern health physics, a number of meaningful dosimetric quantities and units, encompassing aspects of physics, dosimetry and radiobiology, are defined. Table 3.3 outlines the dosimetric quantities and units that are most frequently encountered and figure 3.3 provides a graphical overview of the quantities, as published in ICRP report 103.

An individual may be exposed to radiation from external sources (for example, radiation from nuclear power plants, cosmic rays, CT scans, etc) or to radiation from internal sources (for example tritium in drinking water, inhaled radon, radionuclides used in nuclear medicine procedures, etc). Most dosimetric quantities apply to both external and internal exposure but a few apply only to internal exposure. Internal exposure is complicated by the fact that once committed (ingested) a radionuclide may remain active in the body for a long period of time as a consequence of becoming incorporated into the cellular structure of the body.

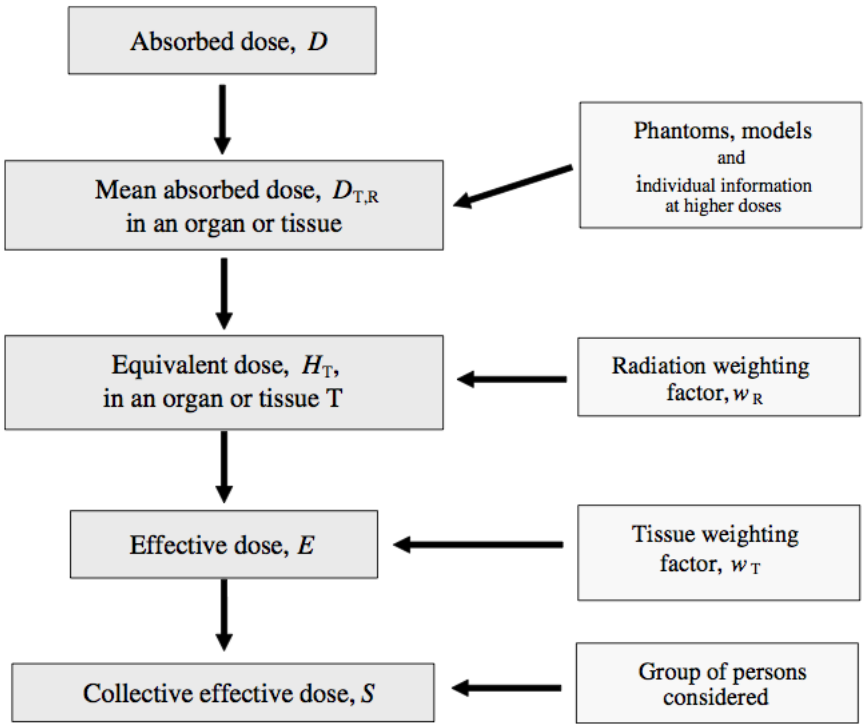


Figure 3.3: The system of dosimetric units as laid out by the ICRP.

Absorbed Dose

The basic physical quantity that is used to measure the "amount" of radiation absorbed in irradiated tissue is the *absorbed dose* D. It is simply a measure of the energy of ionizing radiation absorbed per unit mass of absorbing material.

$$D = \frac{\Delta E_{abs}}{\Delta m}$$

The SI unit of absorbed dose is the gray (Gy), named for Louis Harold Gray; a British medical physicist who studied the biological effects of radiation.

The old unit of absorbed dose is the rad (radiation absorbed dose). The rad is still frequently used in the United States and, as such, may be found in medical/health physics documents and publications written there.

$$1 \text{ rad} = 1 \text{ cGy} = 0.01 \text{ Gy}$$

Table 3.4: Radiation weighting factors W_R, as recommended by the ICRP.

Radiation	Energy (MeV)	W_R
Photons	all	1
Electrons and muons	all	1
Neutrons	0.01 MeV to >20 MeV	5-20
Protons (other than recoil protons)	E ≥2	2
Alpha particles and heavy ions	all	20

Equivalent Dose

Since some radiations are biologically more effective (more dangerous) than others, the ICRP defined the quantity *equivalent dose* H_T which, for a particular tissue or organ T, is the sum of the mean absorbed doses to the organ or tissue as a result of radiations of different types R, each multiplied by an appropriate radiation weighting factor w_{R}.

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

The SI unit of equivalent dose is the sievert (Sv), named for the Swedish medical physicist Rolf Sievert who studied radiation dosimetry and the biological effects of radiation. Table 3.4 lists the ICRP's most recent recommendations for radiation weighting factors (ICRP Publication 103, 2007). The radiation weighting factors are based upon the RBE (Relative Biological Effectiveness, see below) of the various radiations for stochastic effects (see 4.5.2) at low doses and upon the "judgmemt" of the ICRP. Prior to ICRP Report 60, radiation weighting was achieved using so-called Quality factors that were based upon LET rather than RBE.

The old unit of equivalent dose is the rem (röntgen equivalent man). As is the case for the rad, the rem is still in use in the United States.

$$1 \text{ rem} = 1 \text{ cSv} = 0.01 \text{ Sv}$$

Effective Dose

The quantity *effective dose* E was defined by the ICRP to account for the variation in radiation sensitivity among the tissues and organs of the body. The effective dose is defined as the sum of the equivalent doses to exposed tissues and organs multiplied by the appropriate tissue weighting factors w_T.

$$E = \sum_T w_T H_T$$

The sievert is also the unit of effective dose. Table 3.5 lists the ICRP's most recent recommendations for tissue weighting factors (ICRP Publication 103, 2007). The sum of all tissue weighting factors is unity for the whole body

$$\sum_T w_T = 1$$

As is the case for equivalent dose, the old unit of effective dose is the rem.

Organs with higher radio-sensitivity are assigned higher tissue weighting factors. For example, the risk of fatal malignancy per unit equivalent dose is higher for the lung than it is for the thyroid. Indeed, from table 3.5, one could state that following a total body exposure to the body that implied an overall 1.0% risk of cancer, then the risk of cancer to the lung in particular is 12% of 1%, or 0.12%, in comparison to 0.05% for the thyroid.

The effective dose gives a broad indication of the health risk from any exposure to ionizing radiation, regardless of the energy or type of the radiation or the number of organs exposed. It applies equally to external and internal exposure and to uniform and non-uniform radiation.

Collective Equivalent Dose and Collective Effective Dose

In order to compare the equivalent doses (for a particular organ) and the effective doses between exposed population groups, the ICRP introduced the quantities *collective equivalent dose* S_T and *collective effective dose* S.

Table 3.5: Tissue weighting factors W_T, as recommended by the ICRP.

Tissue	$\mathbf{W}_T$
Gonads	0.20
Bone marrow	0.12
Colon	0.12
Lung	0.12
Stomach	0.12
Bladder	0.05
Breast	0.05
Liver	0.05
Oesophagus	0.05
Thyroid	0.05
Bone surface	0.01
Skin	0.01
Remainder of body	0.05
Whole body (sum of all organs)	1.00

The collective equivalent dose S_T is defined as the product of the average equivalent dose to an organ of interest for the members of an exposed population and the number of individuals in the population.

$$S_T = \sum_i \bar{H}_{T,i} N_i$$

The collective effective dose is likewise defined as the product of the average effective dose to the exposed population and the number of persons exposed. The person-sievert is the unit for both the collective equivalent dose and the collective effective dose.

$$S = \sum_i \bar{E}_i N_i$$

Operational Quantities for Area and Individual Monitoring

Equivalent and effective doses are not measurable in practice—it is impossible to accurately quantify the absorbed dose to each organ or tissue that is exposed. They are defined solely for the purposes of radiation protection, e.g., optimisation of procedures and setting of dose limits. Quantities that are used in practice to assess the effective dose to the body or the mean equivalent dose deposited in tissues or organs are known as operational quantities. Dosimeters

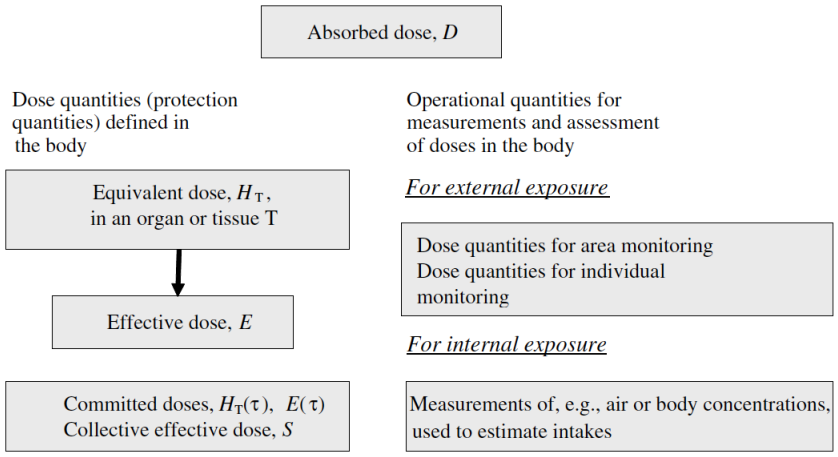


Figure 3.4: The system of operational dosimetric quantities as defined by the ICRU and used in ICRP report 103 (ICRP Publication 103, 2007).

used for radiation protection purposes are calibrated to measure operational quantities. For external exposure, the operational quantities ambient dose equivalent and personal dose equivalent have been defined by the ICRU as shown in Figure 3.4.

In defining the operational quantities (**?**) the ICRU introduced the so-called "ICRU sphere". The ICRU sphere is a simple spherical phantom consisting of tissue-equivalent material and 30 cm in diameter. The properties of the sphere are such that it approximates the human body in terms of scattering and attenuation for a nearby radiation field.

Ambient Dose Equivalent. The ambient dose equivalent $H^*(\mathrm{d})$ is defined as the dose delivered to a depth of d mm in the ICRU sphere under condidtions of broad-beam parallel-field irradiation[1] . The unit of ambient dose equivalent is the sievert. For strongly penetrating radiation a depth of

[1] Broad-beam parallel-field irradiation is seldom encountered in practice. However, to allow for calibrations and measurements it is assumed in practice when using the operational dosimetric quantities.

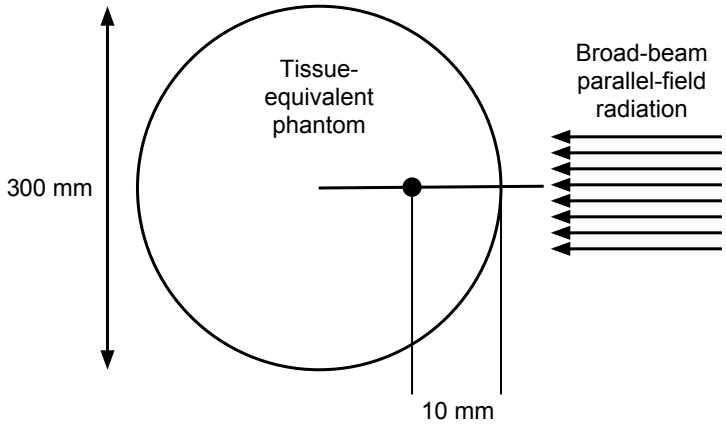


Figure 3.5: The ICRU sphere showing the dose delivery depths used in the definitions of the ambient dose equivalent H*(d) (ICRP Publication 103, 2007).

10 mm is used and for weakly penetrating radiation a depth of 0.07 mm is used, as shown in Figure 3.5.

The ambient dose equivalent is defined such that it provides a measure of the effective dose, or at least a conservative upper estimate on the value of the effective dose for reference person standing in the measured radiation field.

Personal Dose Equivalent. The personal dose equivalent $H_p(d)$ is defined as the equivalent dose in soft tissue below a specified point on the body at an appropriate depth d. The unit of personal dose equivalent is the sievert. The depths used are

- $d = 0.07$ mm below the skin for weakly penetrating radiation
- $d = 3.0$ mm into the eye for weakly penetrating radiation
- $d = 10$ mm below the surface of the body for penetrating radiation

Personal dose equivalents may be measured with detectors worn on the surface of the body that are covered with appropriate thicknesses of tissue equivalent material. Health Canada TLDs are an example of such personal dose equivalent detectors.

3.1.3 Biological Quantities

Some quantities of interest to the health physicist depend on biological factors as well as physical factors. For example, radium-226 has chemical properties similar to calcium. If ingested (as was the case for the radium dial workers, see later in the course) radium may remain in the body for a long period of time. The body treats it like calcium and deposits it in the bones. With its long half-life (1 601 years) and long-term deposition in the body, radium-226 has the potential to cause radiation injury for the rest of the affected person's life.

In health physics, several quantities are defined that take into account both the physical and biological aspects of radiation exposure.

Biological and Effective Half Life

The biological half life of a substance is the time it takes for half of the substance to leave a compartment of a living organism. A compartment may be an organ or a part of the body of interest. The biological decay constant is synonymous with the radioactive decay constant and it gives the probability of a unit substance leaving a biological compartment in unit time.

$$t_{\frac{1}{2}b} = \frac{\ln 2}{\lambda_b}$$

The effective half life of a radionuclide in a compartment of a living organism accounts for both the biological and radioactive half lives of the radionuclide.

Inverse physical and biological half-lives add to give the inverse effective half life

$$\frac{1}{t_{\frac{1}{2}e}} = \frac{1}{t_{\frac{1}{2}p}} + \frac{1}{t_{\frac{1}{2}b}}$$

Physical and biological decay constants add to give the effective decay constant

$$\lambda_e = \lambda_p + \lambda_b$$

Committed Dose

Upon injestion, a radionuclide will deliver dose for as long as it remains a radioactive substance within the host's body—either until the host dies or the substance decays away (physicically or biologically or both), whichever comes first. The commited dose is a quantity that accounts for the total dose delivered to the host due to the injested radionuclide. In calculating the committed dose the effective half-life of the radionuclide in the host's body is used:

$$D_{com} = \int_0^t \dot{D}_0 \times e^{-\lambda_e t}\,\mathrm{d}t$$

where $\dot{D}_0$ is the initial dose rate due to the radionuclide upon injestion, t is the time since injestion and λ_e is the effective half-life of the radionuclide. When integrated, the committed dose is

$$D_{com} = \frac{\dot{D}_0}{\lambda_e}\left[1 - e^{-\lambda t}\right] \tag{3.8}$$

Two extreme situations are easily identified:

After an infinitely long time

When $t \to \infty$ equation 3.8 reduces to

$$D_{com} = \frac{\dot{D}_0}{\lambda_e}$$

For a long effective half-life

For a long t_e, the radionuclide decays little over the time t and so

$$D_{com} = \dot{D}_0 \times t$$

In general t represents the remaining lifespan of the exposed individual. For occupationally-exposed indviduals t is set at 50 years after injestion and for a member of the public t is set up to age 70 years.

The Annual Limit of Intake

The annual limit of intake (ALI) is defined as that quantity of activity of a radionuclide that would lead to the annual dose limit if inhaled or ingested by a "reference person". ALI considers the effective half life of the radionuclide as per the committed dose. It will be discussed later in relation to dose limits.

Relative Biological Effectiveness

The relative biological effectiveness (RBE) is a quantity that is used in radiobiology and health physics to compare the effectiveness of different radiations at producing biological damage. It uses the damage (of a particular type) due to 250 keV X rays, as a normalization factor. RBE for a particular radiation is defined as the ratio of the dose from 250 keV X rays needed to produce a given biological effect to the dose of the particular radiation needed to produce the same biological effect.

$$RBE = \frac{\text{Dose from 250 keV to a tissue}}{\text{Dose from a test radiation for same biological effect}}$$

For example, it only takes one sixteenth of the dose of 14 MeV neutrons to kill cockroach embryos as it does for 250 keV X rays to kill them. Thus, the RBE for cockroach embryo lethality is 16 for 14 MeV neutrons.

The RBE for 250 keV X rays is, by definition, always one for any biological effect.

3.1.4 Legal/Regulatory Quantities

Several quantities are defined for legal/regulatory convenience. They are generally based on physical quantities and are used to set regulatory limits. They

will depend on the jurisdiction where they are defined. Two quantities of interest to health physicists in Canada, the Exemption Quantity and the Transport Index are discussed here.

Exemption Quantity

An exemption quantity (EQ) is defined as the quantity of a radionuclide (expressed in Bq) below which conditions are not imposed by the Canadian Nuclear Safety Commission. Licenses and radiation warning signs are mandated when EQs are exceeded. The following conditions apply:

- For a quantity of radionuclide with an EQ < 1, no special conditions are attached
- For a quantity of radionuclide with an EQ > 1, a license is required
- For a quantity of radionuclide with an EQ > 100, a radiation warning sign is required
- For a quantity of radionuclide with an EQ $> 10\ 000$, written approval (licensing) from the CNSC is required before the radionuclide may be used

Transport Index

The transport index (TI) is defined as the maximum radiation level in mrem/h at one meter from the surface of a single, isolated, undamaged package containing radioactive material.

The definition of the TI can be found in Transport Canada's Emergency Response Guidebook 2008, which is available online. The TI will be discussed again in relation to radiation regulations.

3.2 Radiation Detectors and Dosimeters

Under normal circumstances, humans cannot see, feel, or otherwise detect radiation. As such, we rely on specially-designed instruments to detect and

Table 3.6: The types of radiation detector most commonly used in fields of medical physics and health physics.

Detector type	Signal
Gas-filled counters	Electrical
Thermoluminescent dosimeters (TLDs)	Thermoluminescence
Scintillation detectors	Light
Semiconductor detectors	Electrical
Photographic emulsions	Chemical
Bubble detectors	Bubbles

quantify radiation for us. The purpose of a radiation detector is to produce a signal in proportion to the flux of radiation passing through it. The signal produced depends on the interaction between the radiation and the detection medium, i.e. ionization.

A particle counter is a radiation detector that measures the number (or rate) of ionizing particles that pass through the detector. A radiation dosimeter is a radiation detector that measures, either directly or indirectly, the dosimetric quantities exposure, kerma, absorbed dose or equivalent dose. Dosimeters may provide measurements of accumulated dosimetric quantities or their time derivatives.

In health physics, radiation detectors are used for many purposes. They may be used for routine environmental monitoring, monitoring of occupational exposure, contamination measurements, medical dosimetry, measurements of radon and its progeny and measurements for national security purposes.

Table 3.6 lists the types of radiation detector most commonly used in the medical and health physics fields. Each type is described in detail below.

3.2.1 Gas-Filled Detectors

Gas-filled detectors comprise a volume of insulating gas contained between two electrodes having a voltage difference between them. A schematic of the basic "leaky-capacitor" circuit of a gas-filled detector is shown in figure 3.6. In the absence of ionizing radiation, current cannot flow through the gas and the

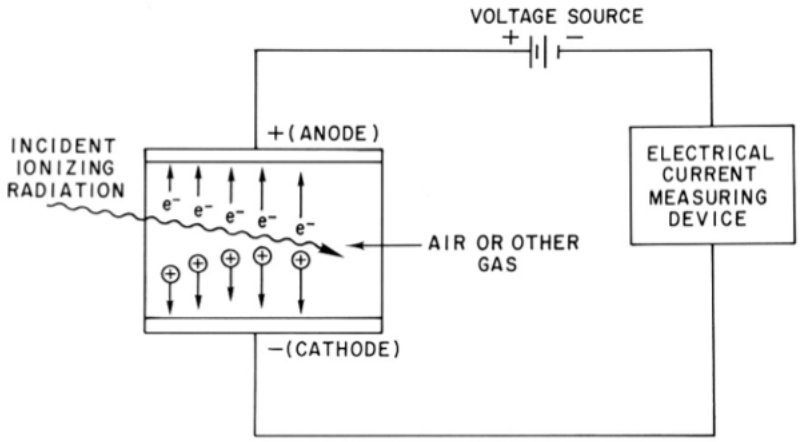


Figure 3.6: Basic leaky-capacitor circuit of a gas-filled detector. Typical voltage is about 250 V.

detector acts as a capacitor. Upon passage of ionizing radiation through the detector, electrons and ions will be produced in the gas and will be attracted to the electrodes resulting in a measurable electrical current.

As shown in Figure 3.7, a gas-filled detector may operate in a number of regions depending on the relationship between the voltage applied and the charge collected[2] . Indeed, gas-filled detectors are categorized according to the voltage region in which they operate.

No detectors operate in the *recombination region* (A) or in the *gas discharge region* (E). In the low-voltage recombination region, the electrical potential is insufficient to collect all charges produced in the gas before they recombine. In the high-voltage gas discharge region, the voltage is sufficiently high that a single ionizing event may initiate a continuous discharge, thereby rendering the detector useless.

[2] Note that no single ionization chamber can be taken through all of the five regions shown in Figure 3.7.

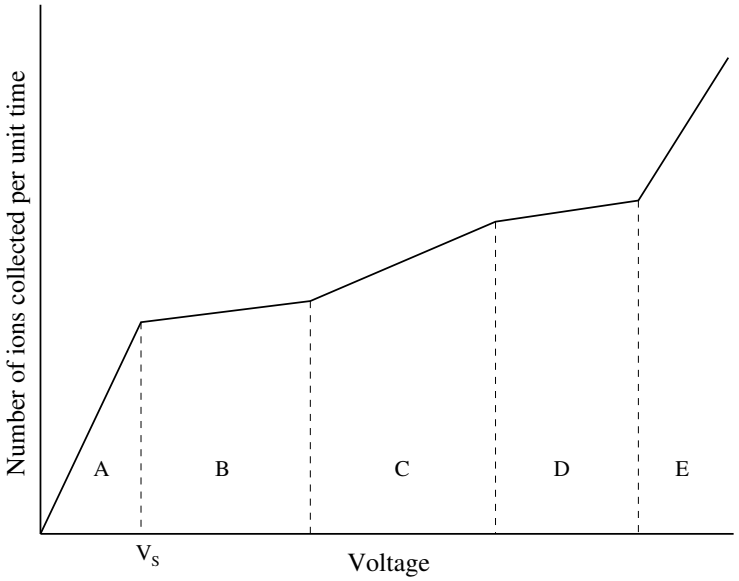


Figure 3.7: The voltage dependence of charge collection in a gas-filled detector. Region A is known as the recombination region. Region B is the ionization chamber or saturation region. Region C is the proportional counter region. Region D is the Geiger-Müller plateau and region E is the continuous discharge region.

Ionization Chambers

At voltages above the saturation voltage V_s, the potential is more than sufficient to collect all charges produced in the chamber's sensitive volume and the detector operates in the *ionization chamber* or *saturation* region (B). No charge multiplication occurs in this region and the charge collected is directly proportional to the number of primary ions (of either sign) produced by the radiation.

Owing to the small electrical signal produced by a radiation event, ionization chambers are seldom used to count single radiation events. Rather, they are typically used to record the total amount of charge (and hence dose) produced by a beam of radiation.

Proportional Counters and Neutron Detectors

For applied voltage beyond the ionization chamber region, a gas-filled detector enters the *proportional counter* region (C), in which the accelerated charges

have sufficient kinetic energy to induce further ionizations through collision interactions. At any given voltage, the output electrical signal is directly proportional to the energy deposited by the primary radiation and primary particle identification, through the use of an appropriate discriminator, is possible. Compared to ionization chambers (for which primary particle identification is also possible, for the same reason), proportional counters produce larger output signals and so require less amplification, making them more practical.

In radiation protection, proportional counters are frequently encountered as neutron detectors. Since primary particle identification is possible, the photon background can easily be discriminated against. The slow neutron signal (from indirect neutron ionization) is enhanced by coating the chamber wall with a boron compound, or by filling the detector volume with BF_3 gas. Alpha particles produced in the $^{10}B(n,\alpha)^7Li$ reaction are easily discriminated from any gamma-ray background. Fast neutrons may be moderated by surrounding the detector with a moderator of hydrogenous material.

Geiger-Müller Detectors

At high voltages, gas-filled detectors operate in the *Geiger-Müller* (GM) region (D). In this plateau region, accelerated electrons excite gas molecules and produce UV radiation. The UV radiation, in turn, induces further ionization such that an avalanche of electron-ion pairs propagates through the gas volume. The output electrical pulse is very strong but is independent of the type or energy of the primary radiation. Accordingly, GM detectors are very sensitive and can detect all forms of ionizing radiation. They are commonly used as survey meters for radiation protection purposes.

Despite their excellent sensitivity, GM detectors have several disadvantages for surveying radiation therapy installations. They are unable to identify particle type since all radiations produce the same signal, and they suffer from

saturation effects when measuring high dose-rate radiation. In the pulsed radiation field of a linear accelerator, the instantaneous dose rate may be extremely high, even if the time-averaged dose rate is considered "normal". This property of linear accelerator beams means that GM detectors are unsuitable for accurate measurements at radiation therapy facilities.

3.2.2 Themoluminescent Dosimeters

TLDs are radiation detectors that are based on the property of thermoluminescence. A thermoluminescent material (often referred to as phosphor) is a crystal with an energy-level diagram similar to that shown in Figure 3.8. Phenomenologically, a thermoluminescent material may be modelled as comprising valence and conduction bands separated by an energy gap, with traps (storage traps and recombination centers) found within the gap. Traps can hold either electrons or positive holes and are generally caused by crystal impurities.

Before irradiation, the traps are empty, i.e., the electron traps are free of electrons and the hole traps contain electrons but no holes. When radiation passes through the crystal, it may excite an electron to the conduction band, from the valence band or from an empty hole trap. A hole is thus left behind in either the valence band or the hole trap. Similarly, a hole may be excited from the conduction band or an electron trap. The system may return to thermal equilibrium via three possible routes: (1) free charge carriers recombine, (2) a free charge carrier recombines with a trapped charge carrier of the opposite sign in a recombination center, with the emission of optical fluorescence, or (3) a free charge carrier becomes trapped in a storage trap and is only released on heating the crystal. When provided with sufficient energy to escape its trap, a charge carrier may move within the valence or conduction band (as appropriate for its charge) until it encounters a recombination trap into which it falls with emission of a visual or ultraviolet (thermoluminescent) photon. Radiation

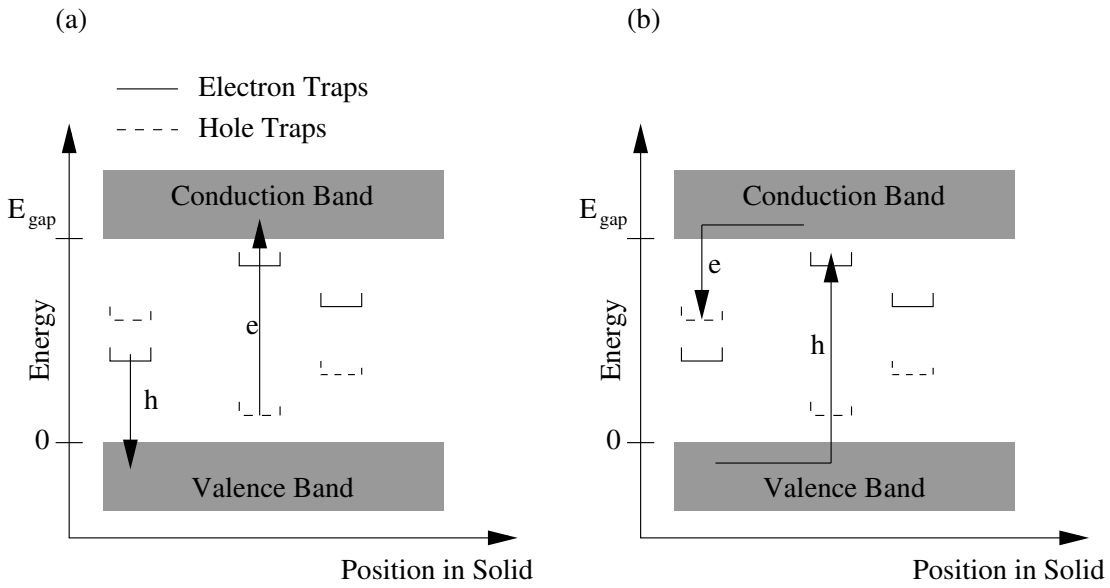


Figure 3.8: Energy-level diagram for a thermoluminescent material, showing electron and hole traps contained within the energy gap between the valence and conduction bands. (a) When irradiated, an electron may be excited from the valence band or a hole trap to the conduction band. Likewise, a hole may be excited from the conduction band or from an electron trap to the valence band. (b) When heated sufficiently, a trapped electron may gain enough energy to escape its trap. It may then move within the conduction band until it encounters a hole trap where it will combine with a hole and emit a visual or UV photon. Likewise, a trapped hole may escape to the valence band and move until it encounters an electron trap.

exposure, controlled heating and light measurement form the process through which TLDs are used to measure radiation.

TLDs are integrating-only (dose accumulation) dosimeters, in that they cannot provide instantaneous dose measurements. However, they offer good sensitivity and a high degree of accuracy in pulsed radiation fields. TLDs are commonly used as personal dosimeter badges, worn by radiation workers. In Canada, a national TLD-reading service is provided by Health Canada. Two important disadvantages of TLDs, are that they are unreliable in low-dose environments due to the faint light signal produced and they can only be read once. The threshold for dose reporting by Health Canada is 0.1 mSv.

3.2.3 Scintillation Detectors

Scintillators are materials, solids, liquids or gases, that absorb ionizing radiation and re-emit it as visible light. To operate, they must be transparent to their own scintillation light. The amount of visible light produced is very small and scintillators are typically coupled to photomultiplier tubes (PMTs) or arrays of photodiodes for efficient light collection and amplification.

Solid (crystal) scintillation detectors (also called phosphors) are made of high-effective-Z material (eg sodium iodide activated with thallium, NaI(Tl)). Because of their high-density and high-Z, solid scintillators are much more sensitive to gamma rays and X rays than gas-filled detectors—ionizing photon interactions are more likely in a crystal than in a gas.

Liquid scintillator detectors are less efficient at producing light than solid scintillators but they are particularly useful for detecting alpha particles and low-energy beta particles that might otherwise suffer from the effects of self-absorption in the source material. By mixing (dissolving or suspending) the source with the detector material itself the charged particles are detected with very high efficiency, close to 100%. Two PMTs are often incorporated into liquid scintillation detectors, as shown in figure 3.9. Since liquid scintillators produce tiny light flashes, having two PMTs operating in coincidence mode allows for discrimination of true light signals against background radiation and electronic noise fluctuations.

Health-physics situations in which one would use a liquid scintillator might include quantification of the amount of tritium in drinking water or the amount of radioactivity in urine. Liquid scintillators are used frequently in nuclear medicine.

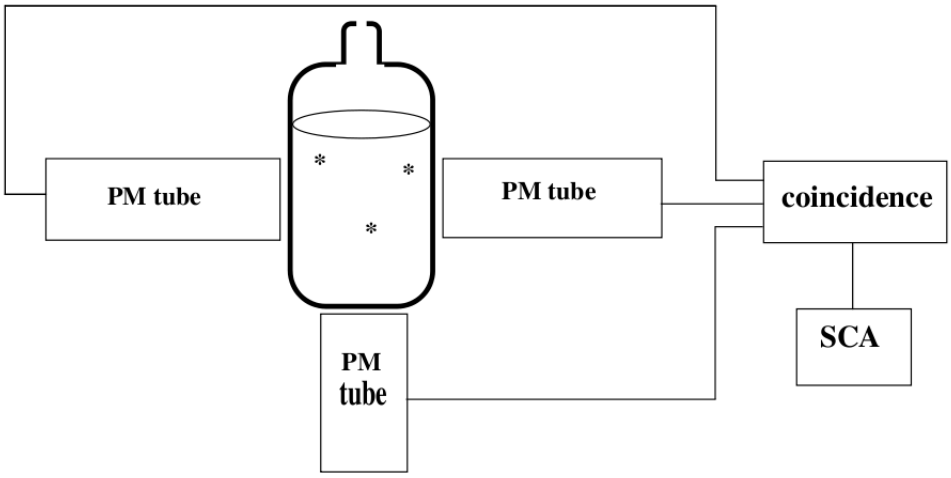


Figure 3.9: Schematic of a liquid scintillation detector with two PMTs operating in coincidence mode. Figure from Cherry et al. (2003).

Scintillators are commonly used as gamma-ray spectrometers since the amount of light emitted is proportional to the energy deposited by the incident ionizing radiation.

Well-type scintillation counters are solid scintillator counters designed such that a test-tube holding a sample radioactive substance may be inserted into a hole in the scintillator. A cross-sectional view of a well-counter detector is shown in figure 3.10. Well counters have excellent geometric efficiency and are typically surrounded by lead shielding to reduce background contamination. They are used frequently in health physics and nuclear medicine.

3.2.4 Semiconductor Detectors

Semiconductor detectors are essentially solid-state ionization chambers—ionizing radiation interacts in the sensitive volume to produce ions than in turn produce a measurable electronic pulse. Compared with gas-filled detectors, however, semiconductor detectors offer greater sensitivity and higher energy resolution. Only $\sim$3.5 eV is required to produce an ion-pair (electron and hole) in a semiconductor, whereas 33.97 eV is required to produce an ion-pair in air. The

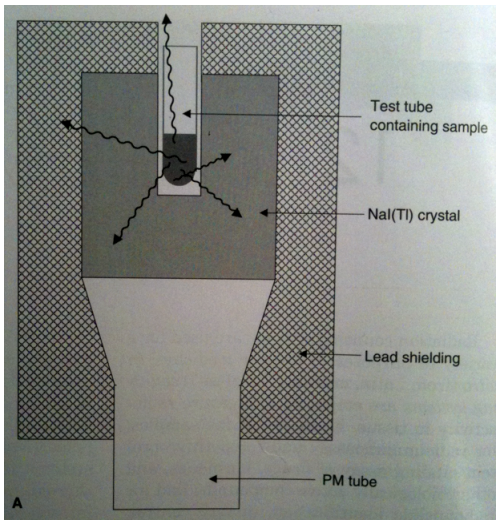


Figure 3.10: Cross-sectional view of a well-type scintillation detector.

energy resolution of semi-conductor detectors is higher than for scintillation detectors ($\sim$1%, compared to $\sim$10%).

Two common materials are used for semiconductor detectors—Si(Li) (silly) and Ge(Li) (gelly). The Ge(Li) detector is more sensitive to thermal noise and must be cooled to liquid nitrogen temperatures. Figure 3.11 is a simple diagram of a semiconductor detector. In health physics, reverse-biased silicon diodes are encountered as electronic personal dosimeters (EPDs). They can detect beta and gamma radiation over a very wide range of doses and dose rates. Since they are electronic, they can integrate, store, and display cumulative dose and produce an alarm if the dose exceeds a certain limit. They can also be connected to a computer for data download. In the United States, the Nuclear Regulatory Commission stipulates that all radiographers and their assistants wear alarming personal dosimeters.

3.2.5 Photographic Emulsion Detectors

Photographic emulsions were the first radiation detectors. In fact, Rötegen discovered X rays by noticing their darkening effect on photographic emulsion.

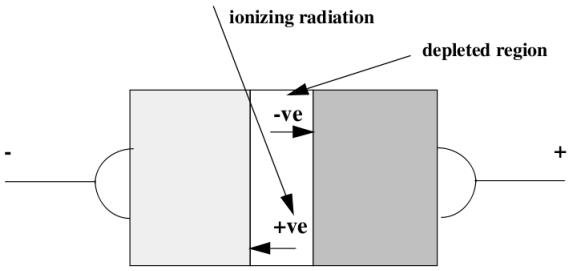


Figure 3.11: Simple overview of a semiconductor radiation detector.

Today, photographic films are on their way out as radiographic detectors, owing to technological advances in digital and computed radiography. However, they are occasionally still encountered in health physics where they may be used for personal dosimetry and for leak testing.

Film badges are essentially badges containing one or more pieces of radiographic film that are worn by radiation workers. Ionizing radiation (alphas, betas, gammas, X rays and neutrons) exposes the silver halides in the emulsion and causes the film to darken. The degree of darkening (optical density) is proportional on the level of radiation exposure. Through appropriate calibration, the film's darkness may be converted into a dose value. The use of filters in film badges allows for determination of the dose from each of the different types of radiation in a mixed radiation environment.

Since films offer a high level of spatial resolution, they are often used as radiation detectors when searching for leaks. For example, one might surround a radiation-producing machine or facility with film to search for inadequacies in the shielding.

3.2.6 Bubble Detectors

Bubble detectors are used to measure neutron exposure. They are passive integrating dosimeters with instant neutron-equivalent-dose readout. A bubble detector consists of a glass tube about the size of a pen filled with thousands

of superheated liquid drops in a stabilizing matrix. When struck by a neutron the potential energy contained within an individual drop is released via a vaporising explosion that produces an audible pop and a visible bubble. The number of bubbles produced is directly proportional to the neutron-equivalent dose H. The bubbles may be counted visually or by a machine. Bubble detectors have a useful lifetime of about three months during which they may be recycled and reused many times. A screw at the end of the tube may be used to re-pressurize the matrix and remove the bubbles.

3.2.7 Calibration of Radiation Detectors

Calibration of radiation detectors used for health physics measurements should be carried out at least once per year by a qualified expert. Calibration is achieved by exposing a detector to a known radiation level and adjusting the detector to read the radiation level. The "known" radiation level should be confirmed using a detector with a calibration of its own traceable to an official standards laboratory, such as the National Research Council (NRC) in Ottawa, Ontario.

Calibration of radiation survey meters is typically performed using some form of shadow-cone technique. Using the shadow-cone technique, a radioactive source with a known exposure rate constant (or air-kerma strength) may be used to calibrate a radiation detector positioned at a measured distance away. The method is illustrated in Figure 3.12. Accounting for the fall-off in exposure as a function of distance, due to the inverse-square law and attenuation in the intervening medium (air), and considering the time over which the source is exposed, the exposure reading on the detector may be predicted. The actual reading on the device, however, will be larger than that predicted, owing to radiation scattered off the surroundings and back to the detector. Although very difficult to predict, the scatter component is easily measured by

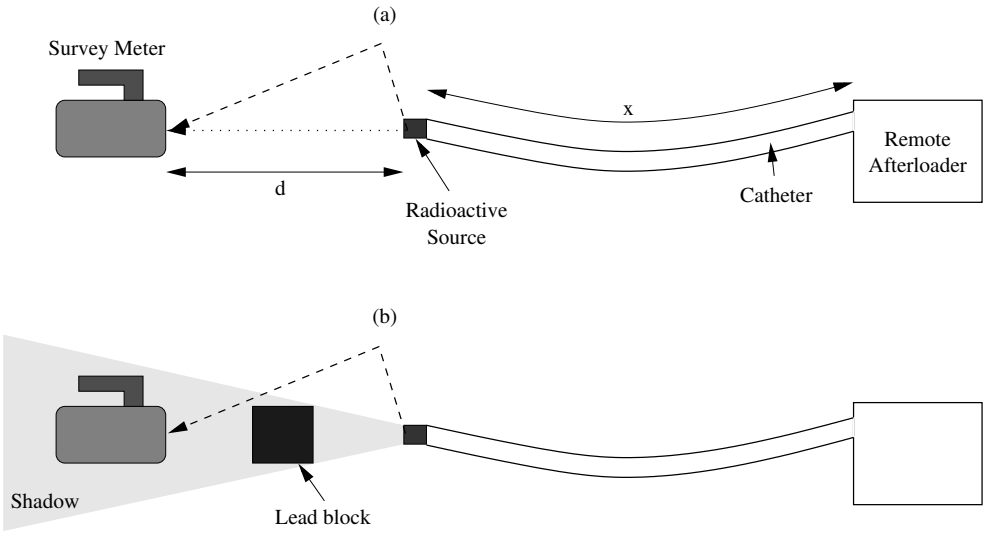


Figure 3.12: Illustration of the shadow-cone technique, as carried out at the MUHC. A survey meter is positioned at distance d from a radioactive source of known air kerma strength. (a) The survey meter is exposed to direct primary radiation (dotted line) and scattered radiation (dashed line) from the source. (b) With a lead block in front of the source, the survey meter is exposed only to the scattered radiation. In both cases, the source travels distance x from the remote afterloader unit to the measurement position through a catheter. The technique facilitates measurement of the primary and scatter components of the radiation reaching the survey meter.

placing a block of attenuating material between source and detector, such that the detector is within the "shadow" of the block and the primary radiation is prevented from reaching the detector. The resulting scatter-subtracted exposure may be compared with the predicted value and hence used to calibrate the detector.

The shadow-cone technique is a nice example of how the radiation level at a point (the detector) depends on distance from the source, the exposure time, the shielding (in this case, attenuation in the air) and scatter. The technique also illustrates the use of the exposure-rate constant.

3.2.8 Radiation Detection in Practice

Radiation detectors come in a variety of flavors. Particle counting detectors measure counts and are useful for measuring half-lives and activities. Solid scintillation counters have high efficiencies for measuring gamma rays and liquid scintillation counters are useful for measuring radiation from liquid sources or weak, self-absorbing, sources. Well counters are used for wipe testing and high-efficiency counting. Survey meters are used when checking for leaks or looking for a source of radiation. Ionization chambers are used for accurate dose measurements. Films are used when spatial resolution is needed, semiconductors and scintillators when energy resolution is important.

Regardless of detector type or function, the following checks should be performed before a detector is used for health physics purposes:

- The detector should be sensitive to the type of radiation to be measured
- The detector must have been calibrated within the last year by a qualified expert
- The detector should pass a battery test, if appropriate
- The detector should pass a rate test (using a pulse generator), if appropriate

- The detector should pass a self-check using a check source

CHAPTER 4
Biological Effects of Radiation

Ionizing radiation has, by definition, the potential to alter the material through which it passes—it may ionize the atoms and molecules it encounters. This is true regardless of the material, be it organic or inorganic, alive or dead. For biological materials, however, ionizing radiation has the particular ability to affect lethal injury.

When absorbed in biological systems, radiation may cause initial injury at the microscopic level. In some cases the microscopic damage may be repaired and go unnoticed, in other cases, it may ultimately manifest itself macroscopically as an observable effect. Macroscopic effects may result from a large radiation exposure that causes sufficient microscopic damage to be noticeable at the macroscopic level, or they may be the propagated long-term consequences of just a single radiation-induced error in the genetic code.

The effects of radiation exposure, particularly at relatively large doses, are well studied. Indeed, the biological effects of radiation exposure are the basis of radiation therapy. In this chapter we will examine the biological effects of radiation and the relationship between dose and radiation-induced injury.

4.1 Sources of Information

Quality data for studies into the effects of radiation exposure are hard to come by. On one hand, data pertaining to very high doses of radiation are rare since very large exposures seldom occur. On the other hand, low-dose data are compromised by a dearth of etiologic (causative) evidence pointing exclusively to radiation. The following sources of information do exist:

- Atomic bomb survivors from Hiroshima and Nagasaki (thousands of people)
- Marshall islanders exposed to fallout from atmospheric nuclear weapons tests (hundreds-thousands of people)
- Victims of accidents at nuclear installations (eg Chernobyl, USSR/Ukraine, 115 cases of acute radiation sickness and 28 deaths)
- Victims of accidents involving medical radiation devices (eg Goiania, Brazil, 48 hospitalizations and 4 deaths)
- Patients exposed to medical radiation for treatment or diagnosis of disease
- Miners exposed to uranium/radon and its decay products
- Residents exposed to radon in the home
- Workers exposed to radium-226 in luminous paint
- Workers exposed to medical radiation
- Animal studies

4.2 The Human Organism

Human beings are organisms. As shown in figure 4.1, we comprise organs that comprise tissues that are made up of cells. Cells consist of cytoplasm containing organelles and a nucleus containing DNA. At the most basic level, cells are made up of protoplasm—organic and inorganic compounds. After all, we are what we eat!

It is our DNA that allows us to survive as individuals and as a species. It enables our cells to divide and multiply (ie growth) and it encodes the genes that we pass to our descendants when we reproduce. Human cells are doubly-coded with DNA (one set of DNA from each parent) and they are either somatic or germ cell. Somatic cells divide by mitosis, germ cells by meiosis.

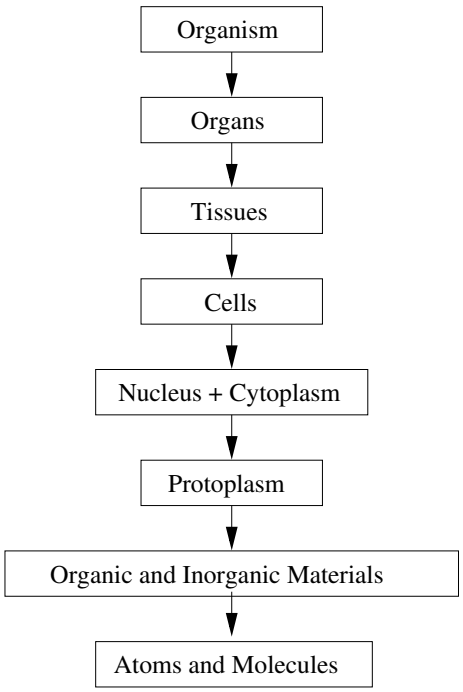
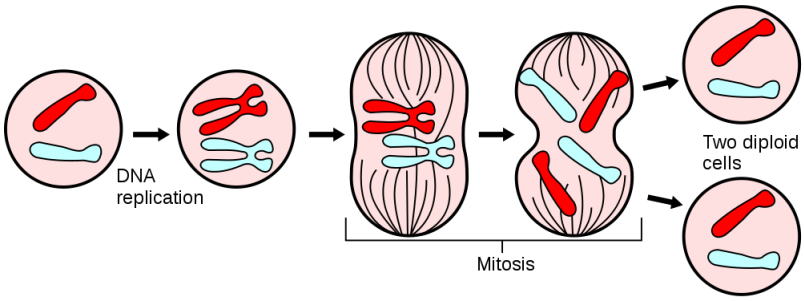


Figure 4.1: The construction of organisms.

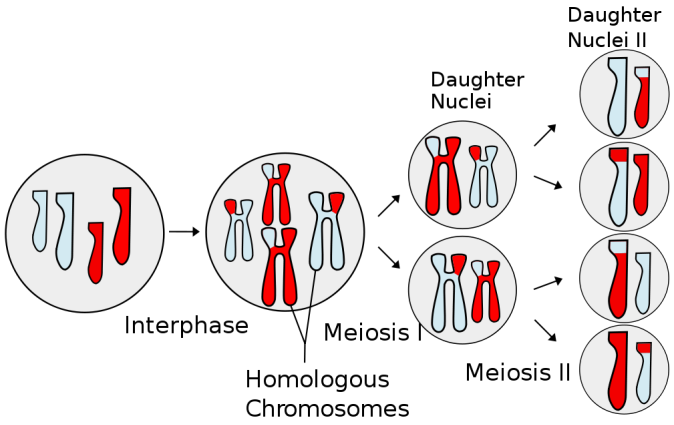
In mitosis the genetic information of one cell is duplicated and then given to two descendant cells that are identical to each other and to the parent cell. In meiosis, the genetic information is duplicated but then spread out amongst four daughter cells. Each daughter cell is different and contains only half the genetic content (DNA) of the parent cell. The daughter cells are known as gametes and are used in sexual reproduction. A zygote is formed when the gametes of each parent combine to, once again, double the genetic information. Figure 4.2 illustrates the processes of mitosis and meiosis.

Our tissues and organs comprise somatic cells. Somatic cells may be categorized as stem cells, transit cells or mature cells.

Stem cells self-perpetuate and produce cells for a differentiated cell population. For example, the hematopoietic system and the lining of the intestine each have stem cells that continuously divide and differentiate



(a) Mitosis



(b) Meiosis

Figure 4.2: Overview of cell division by mitosis and meiosis. (a) In mitosis one parent cell duplicates and splits into two daughter cells. (b) In meiosis, one parent cell duplicates and then splits into four daughter cells.

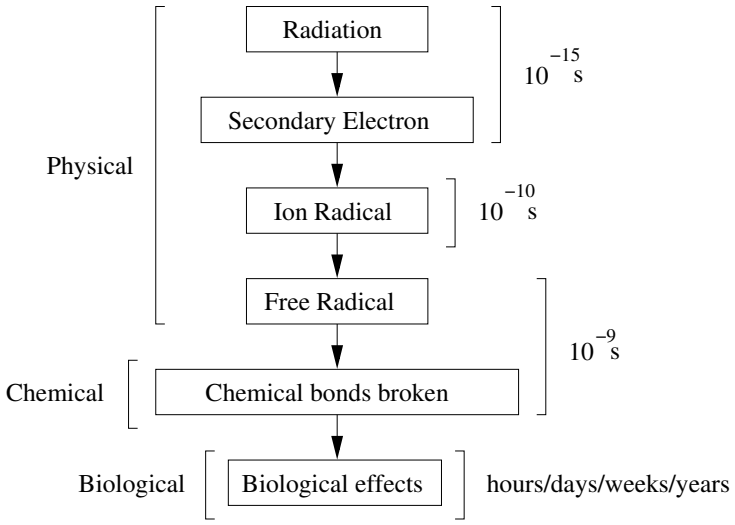


Figure 4.3: Overview of sequence of radiation damage.

into the cells that are needed to replenish the blood supply and the lining of the intestine, respectively.

Transit cells are cells that are in movement to another population. For example, immature red blood cells develop in bone marrow and become mature red blood cells after about a day of actively circulating within the blood supply of the body.

Mature cells are cells that are fully differentiated and no longer divide. For example, muscle cells and brain cells are fully developed and do not further reproduce.

4.3 Sequence of Radiation Damage

The initial effects of ionizing radiation (ie ionization) are physical and immediate. The physical effects (the production of ions) may result in rapid chemical changes at the molecular level. The chemical changes may, in turn, result in biological damage, to either the DNA (mainly) or to other cellular components. Figures 4.3 and 4.4 present overviews of the damage sequence and the timescale of injury.

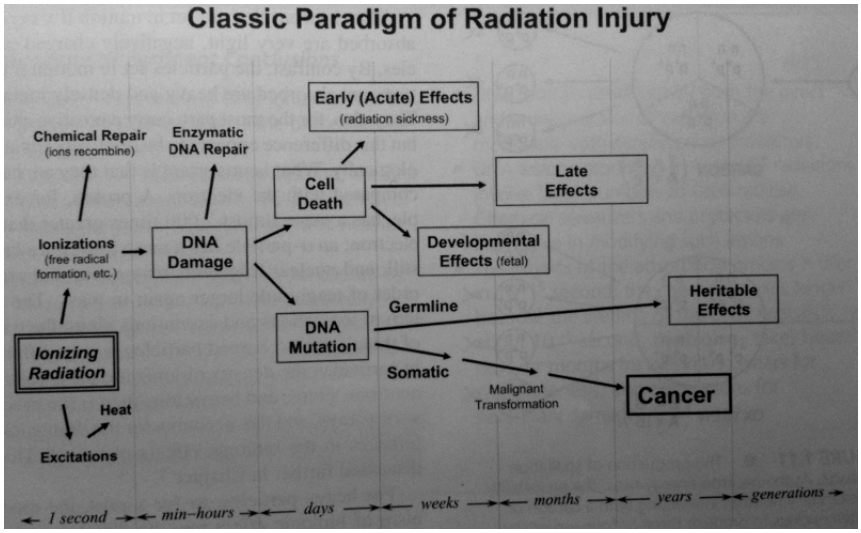


Figure 4.4: Overview of sequence of radiation damage. Figure from Hall and Giaccia (2006).

4.4 Radiation Damage at the Cellular Level

4.4.1 Direct and Indirect Action in Cell Damage by Radiation

Radiation incident on a cell may directly or indirectly damage the DNA of the cell.

Directly acting radiation is ionizing radiation that interacts with and damages the cell's DNA without recourse to reactive chemical agents.

Indirectly acting radiation is ionizing radiation that interacts first with the protoplasm of the cell to produce reactive chemical agents (free radicals), which in turn react with and damage the cell's DNA.

Figure 4.5 illustrates the processes of direct action and indirect action for radiation damage at the cellular level.

Direct action is the dominant interaction process for high-LET (densely ionizing) radiation, since the probability of the radiation interacting with the small DNA target is high. Conversely, indirect action is favoured for low-LET (sparsely ionizing) radiation.

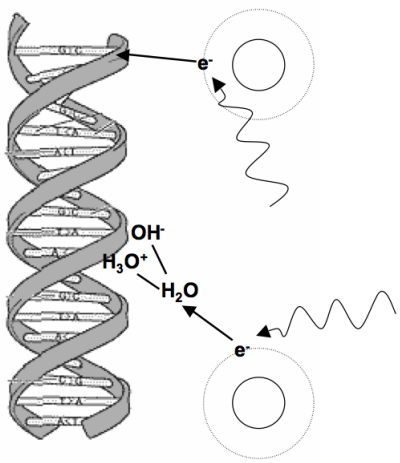


Figure 4.5: Illustration of the direct (top) and indirect (bottom) action of radiation in damaging DNA.

4.4.2 DNA Strand Breaks

Human DNA has a double-strand helical structure, whereby two complementary strands (molecular chains) of DNA are joined together and wrapped around each other in the form of a double helix. Figure 4.5 includes a sketch of a segment of human DNA. Radiation damage to the DNA double helix may take the form of either a single or double-strand break. Double strand breaks are more dangerous since they are very difficult (if not impossible) to repair. Single strand breaks, on the other hand, may be repaired owing to the information redundancy encoded into the surviving DNA strand.

Double-strand DNA breaks may be caused by the effects (ie direct action or indirect action) of all ionizing radiations. However, they are more likely to occur as the result of high-LET radiation (and hence by direct action), where the probability of successive ionization events coincides with the dimensions of the DNA double helix width ($\sim$2 nm). The effect of LET on radiation damage is most clearly borne out in a plot of RBE against LET, as shown in figure 4.6. The RBE peak occurs at an LET of $\sim$100 keV/μm, which at about one ionization per 2 nm, roughly corresponds to the width of the DNA double-helix

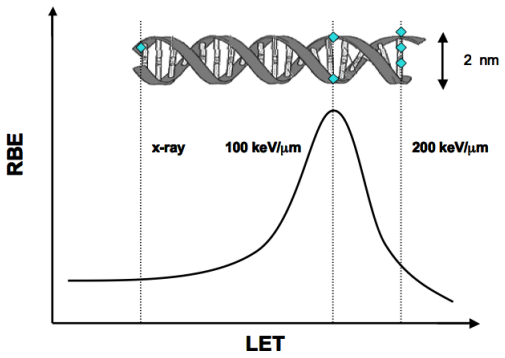


Figure 4.6: The relationship between LET and RBE and an illustration of the reason for it.

and hence an increased probability of an efficient double-strand break. Beyond ∼100 keV/μm the RBE falls off due to high LET radiation effectively having wasted dose—the double-strand break has already been produced and the increase in dose dose not cause an increase in biological damage. Remember, RBE is a ratio of doses for a given biological effect (in this case, for a double-strand break).

4.4.3 Fate of Irradiated Cells

Irradiation may result in nine possible consequences for the cell. They are:

1. No effect
2. Delayed cellular division
3. Apoptosis—programmed cell death (as per leaves falling from the trees in the Fall)
4. Reproductive failure—immediate inability to divide mitotically
5. Genomic instability—eventual reproductive failure after a number of divisions
6. Mutation—cell survives but with a mutation
7. Transformation—cell is changed biologically but survives (may lead to a change in the individual or carcinogenesis)

8. Bystander effects—the irradiated cell sends signal to neighbouring unirradiated cells that induces genetic damage in them
9. Adaptive responses—the irradiated cell adapts to radiation and becomes more resistant to subsequent radiation

4.5 Radiation Injury at the Macroscopic Level

Biological damage to a large number of cells may be significant enough to be macroscopically noticeable in the short to medium term (hours to days). This is particularly true in the case of damage to stem cells that may result in an inability to replenish a needed supply of mature cells. For example, the hematopoietic syndrome (described below) becomes noticeable in an individual about 3 weeks after exposure as mature circulating blood cells die off but are not replaced due to the depletion of hematopoietic stem cells.

Biological damage to just one or a few cells is not immediately noticeable at the macroscopic level. However, even a small amount of DNA damage, if propagated through cell division, may result in long-term effects. Such long-term effects may include cancer if somatic cells are injured, or sterility/generational effects if germ cells are injured.

Depending on the number of cells affected, the macroscopic biological effects of radiation can be divided into deterministic and stochastic effects. Deterministic effects are generally (but not always) the result of large radiation doses in which many cells are simultaneously injured. Stochastic effects involve generally (but, again, not always) smaller radiation doses, few cells, and entail radiation-induced carcinogenesis.

In all cases, the likelihood of a macroscopic biological effect occurring increases with increasing microscopic damage. Table 4.1 lists and contrasts the properties of deterministic and stochastic effects of radiation.

Table 4.1: Deterministic and stochastic effects of radiation damage.

Property	Deterministic Effects	Stochastic Effects
Threshold dose	Yes	No
Probability	Independent of dose (above threshold)	Proportional to dose
Severity	Proportional to dose	Independent of dose
Applicable to	Individuals	Populations
Expression	Generally early effects	Generally late effects/cancer
Dose level	Typically involve large doses	Can involve any dose

4.5.1 Deterministic Effects

Deterministic effects of radiation are biological effects that increase in severity with increasing absorbed dose in an exposed individual. They have dose thresholds above which the effect will definitely occur, and below which it will not, as shown in Figure 4.7a. The threshold is specific to the exposed individual but average thresholds may be determined for a population. Examples of deterministic effects include radiation cataractogenesis, tissue fibrosis, organ atrophy (wasting away) and the syndromes of whole body radiation sickness.

Deterministic effects also often occur as early effects of radiation damage and they usually (but not always) involve relatively large radiation doses.

$\mathbf{LD_{50}}$

In examining dose-response characteristics for radiation exposure, the 50% dose is often used. It corresponds to the dose at which 50% of the exposed population responds in the expected way. When lethality (death) is the biological end point of interest, the dose is referred to as the LD_{50} dose, or the dose at which death is expected for 50% of the exposed population. The $LD_{50/30}$ and $LD_{50/60}$ quantities are often used and refer to the lethal dose for 50% of the exposed population in 30 days or 60 days, respectively.

Syndrome	Threshold Dose	Description
Hematopoietic syndrome	2.5 Gy to 5 Gy	Characterized by death of bone marrow due to sterilization of the bone marrow precursor cells. Without major intervention, the affected individual will die within weeks or months.
Gastrointestinal syndrome	5 Gy to 12 Gy	Involves depletion of the stem cells that generate the epithelium of the gut. Death is expected within days.
Cerebrovascular syndrome	Above 100 Gy	Involves breakdown of the neurological and cardiovascular systems. Prognosis is hopeless and death occurs within hours of exposure.

Table 4.2: The three whole-body radiation sickness syndromes.

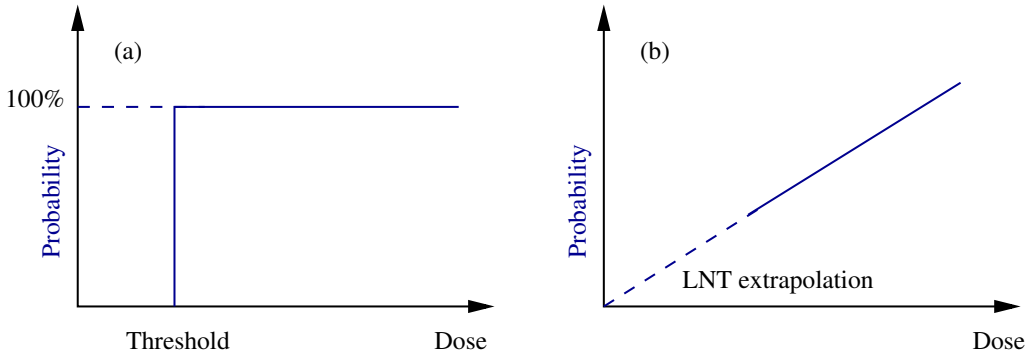


Figure 4.7: The probability of the biological effects of radiation against dose. (a) Deterministic effects have a threshold dose beyond which the probability of occurrence is 100% (b) Stochastic effects have no threshold. In the LNT model, the observed linear dose-dependence at high doses is extrapolated back to the origin.

Acute Whole Body Radiation Sickness

Acute whole body radiation sickness (also referred to as acute radiation syndrome, ARS) occurs following acute high dose whole-body exposures. An immediate prodromal syndrome followed by one of three main syndromes are observed: the hematopoietic syndrome, the gastrointestinal syndrome and the cerebrovascular syndrome (blood, gut and brain). Each of the three main syndromes is characterized by the dose range involved and the part of the body affected. Table 4.2 provides a brief summary of the three whole-body radiation sickness syndromes. More details are given below.

The **prodromal** (or early) radiation syndrome occurs after exposure of a few gray. It represents the body's initial response to the large dose of radiation

and it is a precursor of one of the three main syndromes. The prodromal syndrome is characterized by anorexia (loss of appetite and inability to eat), nausea, vomiting and fatigue.

The **hematopoietic** syndrome occurs following radiation exposure in the range of 2.5 Gy to 5 Gy. The main symptoms of the syndrome include chills, fatigue, mouth ulcers and a suppressed immune system. Hematopoietic stems cells (the precursors of red and white blood cells and platelets) are sterilized during the radiation exposure and consequently cannot provide an adequate fresh supply of blood cells to replace mature circulating cells as they die off. Since mature cells circulate in the body for several weeks, the hematopoietic syndrome has a latency period of this duration. Without treatment, in the form of a blood transfusion and antibiotics, death due to infection is the likely outcome since the body's immune system is destroyed.

In medicine, the hematopoietic syndrome is sometimes induced in leukaemia or myleoma patients who will undergo a bone marrow transplant. By using total body irradiation (TBI) with doses on the order of 10 to 12 Gy in four fractions, the patient's immune system can be sufficiently suppressed that it won't reject the donor's bone marrow. To prevent infection post-irradiation, the patient requires very careful nursing in a sterile environment and the use of antibiotics.

The **gastrointestinal** (GI) syndrome occurs following whole body doses of greater than 10 Gy. Such doses are more than sufficient to cause the hematopoietic syndrome but because of the latency period for hematopoietic symptoms, the GI syndrome is observed first. People experiencing the GI syndrome die from their exposure within days. The GI syndrome is characterized by nausea, vomiting and prolonged diarrhoea. The symptoms may be attributed to depopulation of the epithelial lining of the gastrointestinal tract

by the radiation. Like the blood system, the lining of the intestine comprises differentiated cells with a natural life cycle that are continuously replaced by cells produced from stem cells. As is the case for the hematopoietic syndrome, the sterilization of gastrointestinal stem cells leaves the intestinal lining unreplenishable. Autopsies of people who experienced the GI syndrome have shown that their intestinal linings were lost.

The **cerebrovascular** syndrome occurs in people who are subjected to huge radiation exposures exceeding 100 Gy. Death occurs within hours. The exact causes of the syndrome are unclear since it happens so rarely. All organs are seriously damaged but it appears that death is due to damage to the brain and the nervous system, perhaps due to swelling within the skull. Severe nausea and vomiting are observed within minutes, followed by disorientation, convulsions, coma and finally death.

4.5.2 Stochastic Effects

Stochastic effects are biological effects in which the risk of the effect occurring, rather than its severity, increases with radiation dose, as shown in Figure 4.7b. Stochastic effects include radiation carcinogenesis and genetic effects. Stochastic effects are probabilistic, do not have a threshold level, and apply to populations rather than individuals.

The *linear-no-threshold* (LNT) model is a model of the dose-risk relationship for stochastic biological effects. It makes an assumption that the linear increase in the probability of a stochastic effect, seen at high doses, can be extrapolated back to the low dose regime, indeed to the origin, as shown in Figure 4.7b. The no-threshold premise of the LNT model is controversial in that data relating to radiation effects at low doses are statistically compromised by low frequency and the presence of naturally-occurring spontaneous carcinogenesis. However, it is a usefully conservative model for use in radiation

protection—it suggests that no dose level is safe and that exposure to ionizing radiation should be both minimized and justified.

The ALARA Principle

The ALARA principle is central to the concept of radiation protection. It stipulates that all/any exposure to ionizing radiation should be kept As Low As Reasonably Achievable, economic and social factors being taken into account—essentially a manifestation of the LNT model. In practical terms, it means that if it is possible, within reason (ie the economic and social benefits of the radiation accounted for), to lower the radiation exposure of an individual, then it should be lowered. The use of radiation should not imply exposure to individual workers or members of the public that approaches the regulatory limits. Rather, every effort should be made to ensure that the dose limits are never reached.

Cancer

Cancer (malignant neoplasm) is a class of many diseases in which the cells within a tissue of an organism undergo uncontrolled division, intrude on nearby tissues, and sometimes metastasize (spread) to distant tissues. Most cancers form neoplastic tumors (lesions of cancer cells) but some, such as leukaemia, do not. Benign tumors are self-limited tumors that neither invade nor metastasize.

As described earlier, human cells are either somatic or germline and they propagate by cell division. Somatic cells make up the tissues and organs of the body, whereas germ cells make up the gametes (spermatozoa and ova). Normal healthy tissue comprises somatic cells in homoeostasis—cell creation and cell death in equilibrium. Maintenance of homoeostasis is dependent upon regulated cell division (mitosis) and programmed cell death (apotosis). Any

interruption in the regulation process may result in malignant progression from homoeostasis to metastasis and thus the development of cancer. Three groups of genes, proto-oncogenes, tumor-suppressor genes and DNA stability genes, regulate the division and death of somatic cells.

It is at the DNA-level that carcinogenesis manifests itself: DNA damage within the three groups of control genes may permit uncontrolled growth and cellular immortality. Carcinogenesis may occur spontaneously through random errors in DNA replication or may result from exposure to carcinogenic agents such as chemical mutagens, ionizing radiation, UV and viruses.

Cancer may affect people (and animals) of all ages, although the risk for most types of cancer increases with age. It is estimated (Canadian Cancer Society, 2009) that 40% of Canadian women and 45% of Canadian men will develop some form of cancer during their lifetime. According to the Canadian Cancer Society, 75 300 deaths due to cancer were expected in Canada during 2009. Indeed, the current cancer incidence in developed countries is about 5 000 new cancers per one million population, with the rate steadily increasing by some 3% per year.

4.5.3 Effects of In-utero Irradiation

The developing embryo is highly sensitive to ionizing radiation due to extensive cell division and rapid growth. Gestation may be divided into three main stages: pre-implantation (up to 9 days following conception), major organogenesis (2nd to 8th week after conception) and fetal growth (end of major organogenesis until term). Each stage is characterized by a different response to radiation.

At the pre-implantation stage, the conceptus, which is still a small number of undifferentiated cells, is generally affected in an all-or-nothing manner. Either the radiation causes fatal damage, or the cells survive. During major

organogenesis, cellular differentiation and the formation of tissues and organs occurs. Each organ and tissue is most sensitive at the peak of its differentiation. The CNS appears to be particularly sensitive during major organogenesis and microcephaly was observed in the children born by Hiroshima survivors who were exposed at this stage. The final stage of development, fetal growth, appears to be the least sensitive to radiation, although behavioural alterations and reduced intelligence later in life have been observed. The biological effects of prenatal radiation were studied by the ICRP and reported in publication 90 (ICRP Publication 90, 2003).

CHAPTER 5
Radiation Protection Organisations

As we have seen, radiation offers great benefits to society but presents significant health risks. It is an toxic agent undetectable to our senses, the deterministic and stochastic consequences of which are delayed, such that our repulsive reflexes cannot warn and protect us. To mitigate the, non-obvious, risks, public health policy mandates maximum radiation dose limits for non-background exposure. Two sets of limits are imposed; one set for members of the public and another, less restrictive, set for people who are likely to become exposed at work and who accept the higher risk posed by their occupation.

In this chapter we will review the organizations that recommend and set dose limits for radiation exposure.

5.1 Historical Perspective

The science of radiation protection (ie health physics) grew in parallel to developments in diagnostic radiology and radiation therapy, following the discovery of radiation by Röntgen in 1895. The earliest recorded biologic effects of radiation were reported within a year of Röntgen's discovery and included skin "burns", epilation and eye irritation (Hall and Giaccia, 2006). In 1915, the British Röntgen Society introduced the first formal proposals for radiation protection. A little over a decade later, in 1928, the British proposals were internationally adopted at the Second International Congress of Radiology in Stockholm. At the same congress, the International X-Ray and Radium Protection Committee (IXRPC) was established. Following World War II, and the renewed urgency for radiation protection precipitated by the nuclear arms

race, the IXRPC was reconstituted as two separate commissions: the International Commission on Radiological Protection (ICRP) and the International Commission on Radiation Units and Measurements (ICRU), both of which still operate.

The US equivalent of the IXRPC was the Advisory Committee on X-Ray and Radium Protection, which operated from 1929 until the end of World War II. In 1946, it was replaced by the National Council on Radiation Protection and Measurements (NCRP), a non-governmental public service organization with a congressional charter to provide recommendations regarding radiation protection. NCRP reports form the basis of radiation protection policy within the United States and are consulted internationally for their scientific value.

In Canada, the Atomic Energy Control Board (AECB) was established under the Atomic Energy Control Act of 1946, with a charge to assist the Canadian government in matters pertaining to radiation in regulation, mining and research. In 2000, the Canadian Nuclear Safety Commission (CNSC) replaced the AECB. The commission's mandate is stipulated in the Nuclear Safety and Control Act of 1997 and it serves to regulate the use of nuclear energy and materials in Canada.

5.2 Modern Organizations for Radiation Protection

Today, numerous organizations are involved in matters pertaining to radiation protection. Table 5.1 provides a list of the most important organizations with a summary of the services they provide. All levels of organization are encountered; international, national, provincial/state, municipal and institutional.

Table 5.1: A list of international and national organizations involved in radiation protection.

Acronym	Name	Jurisdiction	Reports to	Activities	Notable Reports
UNSCEAR	United Nations Scientific Committee on the Effects of Atomic Radiation	International	UN General Assembly	Advisory (academic/scholarly)	
BEIR	Biological Effects of Ionizing Radiation	US but with international representation	US National Academy of Sciences (ultimately reports to the US Congress)	Advisory (academic/scholarly)	
ICRP	International Commission on Radiological Protection	International	Medical/health physics community	Advisory (recommending)	23 (Reference man), 84 (Pregnancy and medical radiation), 103 (Recommendations of 2007 - including dose limits)
ICRU	International Commission on Radiation Units and Measurements	International	Medical/health physics community	Advisory (recommending)	37 (Stopping power)
IRPA	International Radiation Protection Association	International	Medical/health physics community	Facilitates communication	
IAEA	International Atomic Energy Agency	International	UN General Assembly (and, when appropriate, the UN Security Council)	Safety and security, science and technology. Practical assistance to developing countries	SRS 47 - Shielding for Radiation Therapy
WHO	World Health Organization	International	United Nations	Sets standards for healthcare, examines health trends, technical support	
IEC	International Electrotechnical Commission	International	Electrical/electronic engineering community	Sets international standards for electrical/electronic devices	
NCRP	National Council for Radiation Protection	National (US)	US Congress	Advisory (recommending)	49 (Radiation therapy shielding up to 10 MV), 86 (Biological effects of RF radiation), 147 (Diagnostic radiology shielding), 151 (Radiation therapy shielding), 160 (Exposure to the population)
NIST	National Institute of Standards and Technology	National (US)	US scientific community	Advisory (academic/scholarly)	Reference data on the web
Canada NRC	Canada National Research Council	National (Canada)	Canadian scientific community	Advisory (academic/scholarly). Primary standards laboratory in Canada	
CNSC	Canadian Nuclear Safety Commission	National (Canada)	Canadian government	Regulatory body for the use of nuclear radiation in Canada	Authority vested in the CNSC by the Nuclear Safety and Control Act (1997)
US NRC	US Nuclear Regulatory Commission	National (US)	US government	Regulatory body for the use of nuclear radiation in the US	
HC	Health Canada	National (Canada)	Canadian government	Advisory/regulatory, national dosimetry service	Radiation Emitting Devices Act (1985)
TC	Transport Canada	National (Canada)	Canadian government	Advisory/regulatory	Transport of Dangerous Goods Act (1992)

5.2.1 International

Organizations interested in radiation protection at the international level are advisory in statute. They may provide recommendations based on their scientific findings (for example, the ICRP and the IAEA) or they may simply present scholarly reports of their scientific investigations (for example, UNSCEAR). A number of international organizations (for example the IAEA, the WHO and the IRPA) also provide practical scientific support for member countries, particularly for developing countries that do not have well-established national radiation protection organizations of their own.

5.2.2 National/Provincial

At the national level, radiation protection organizations may provide non-binding advice and/or scientific analysis or they may draft nationally binding regulations. Examples include the CNSC, Health Canada's Radiation Protection Bureau and Transport Canada in Canada and the NCRP and NRC in the United States. Provincial/state organizations typically draft binding regulations that have jurisdiction over areas of radiation protection not covered by national regulations.

In Canada, radioactive materials (including all nuclear substances, nuclear medicine and high-energy radiation therapy, which involves induced-radioactivity) are regulated at the federal level by the CNSC, while x-ray production (diagnostic radiology) is the responsibility of the provinces. In Quebec, regulations pertaining to the use of radiation-emitting devices were drawn up by the Service de Radioprotection of the Ministère de l'Environnement and are stipulated in the Quebec civil code.

5.2.3 Municipal/Institutional

At the municipal and institutional levels, radiation protection is generally a matter of solicitation and involves radiation safety committees established for the purpose of coordinating radiation protection programs in accordance with provincial/national regulations. For example, two radiation safety officers (physicists) are employed at the McGill University Health Centre, one with responsibility over radiation safety in the department of Radiation Oncology and the other with general responsibility for radiation safety throughout the hospital.

CHAPTER 6
Radiation Protection Regulations

In Canada, the CNSC regulates radiation dose limits and the production and use of radioactive materials using the authority vested in it by the "Nuclear Safety and Control Act" (NSCA) of 1997. The import/export and sale of radiation-emitting devices (including diagnostic x-ray equipment and x-ray imaging machines for airport security) are regulated by Health Canada under the "Radiation Emitting Devices Act" (RED Act) of 1985. The transport of radioactive material is regulated by Transport Canada according to the "Transport of Dangerous Goods Act" (TDG Act) of 1992.

6.0.1 The Nuclear Safety and Control Act

The NSCA was approved by the Canadian Parliament in 1997 and came into force in May 2000. It replaced the Atomic Energy Control Act of 1946 and replaced the Atomic Energy Control Board with the, more powerful, CNSC, as the sole authority to regulate the production and use of radioactive materials in Canada. CNSC jurisdiction covers the following uses of radiation:

- Power reactors
- Non-power reactors
- Nuclear research and test establishments
- Uranium mines and mills
- Processing and fuel fabrication facilities
- Heavy water production plants
- Nuclear substance processing facilities
- Particle accelerators

- Waste management facilities
- Packaging and transportation of nuclear substances
- Nuclear substances and radiation devices
- Lands under evaluation
- Irradiators
- Imports and exports of nuclear items
- Exports of nuclear-related dual-use items
- Dosimetry service providers

The NSCA provides the legal definition of a "nuclear energy worker" (NEW) in Canada. A NEW is defined as

> A person who is required, in the course of the person's business or occupation in connection with a nuclear substance or nuclear facility, to perform duties in such circumstances that there is a reasonable probability that the person may receive a dose of radiation that is greater than the prescribed limit for the general public.

The NSCA also provides the legal definitions for nuclear facility, nuclear substance, prescribed equipment (ie equipment that is controlled by the act) and radiation, amongst other terms.

The main purpose of the NSCA is to provide a legal basis and framework of operation for the CNSC, which was established under the act. The makeup of the CNSC and the powers vested in it are detailed within the act. Notable powers of the commission include the power to act as a court of record, the power to establish classes of licences for the use of radioactive materials within Canada, the power of inspection and the power to search without warrant in order to ensure compliance with the NSCA. The CNSC is also provided with the power to regulate the use of radioactive materials, the use of prescribed

equipment, the disclosure of prescribed information, the setting of radiation dose limits and the protection of nuclear energy workers, amongst other details.

6.0.2 CNSC Regulations of Interest

In this section we will examine several of the CNSC regulations and highlight certain points of interest relevant to this course in Health Physics. The regulations and the NSCA itself are quite readable and may be downloaded from the CNSC website (www.cnsc-ccsn.gc.ca).

Radiation Protection Regulations

The radiation protection regulations apply generally to all licensed uses of radiation covered by the NSCA, except to the medical administration of radioactive substances to a patient or to a biomedical research volunteer (for which there is a sole requirement to advise the person, to whom the radioactive substance has been administered, of the methods by which he/she may reduce the exposure to others upon leaving the hospital/clinic). Some highlights of the radiation protection regulations include:

- The requirement to put in place a radiation protection program when using CNSC-licensed substances/equipment.
- The requirement to keep effective and equivalent doses ALARA, social and economic factors being taken into account.
- The requirement to monitor and record effective and equivalent doses.
- The definition of an "action level", as the level of radiation dose that, if reached, may indicate loss of control of the radiation protection program.
  - the action level is set, as appropriate, in each individual license
- The requirement to undertake the following actions should an action level be reached:
  - investigate the cause

  - fix the problem
  - notify the CNSC within the time period specified in the licence
- The setting of dose limits from man-made radiation in Canada.
  - the Canadian limits on effective dose and equivalent doses, as set by the CNSC, are listed in table 6.1
- The requirement to inform nuclear energy workers, in writing, that they are nuclear energy workers, explain to them the associated risks and dose limits and to provide them with their radiation dose levels.
- The requirement to obtain from nuclear energy workers written acknowledgement (i.e., signature) that he/she has received the information.
- The requirement to inform each female nuclear energy worker, in writing, of the rights and obligations of pregnant nuclear energy workers.
  - the licensee must make any accommodation, that does not hurt his/her business, to ensure that the pregnant nuclear energy worker does not exceed the dose limits set for pregnant nuclear energy workers.
  - a pregnant nuclear energy worker is required to inform the licensee, in writing, immediately when she becomes aware that she is pregnant
- The requirement to use a licensed dosimetry service (e.g, the National Dosimetry Service of Health Canada) to measure and monitor the doses of radiation recived by and committed to nuclear energy workers who have a reasonable probability of recieving an effective dose greater than 5 mSv in a one-year dosimetry period.
  - The licensed dosimetry service is in turn required to file all measured and monitored doses with the National Dose Registry that is maintained by Health Canada.

- During an emergency, the dose limits for non-pregnant nuclear energy workers, are increased to 500 mSv effective dose and 5 000 mSv equivalent dose (essentially a 10-fold increase on the non-emergency limits).
  - Since the limits for pregnant nuclear energy workers are not increased during an emergency, it is common practisee to remove pregnant nuclear energy workers from work situations in which emergencies may occur. For example, at the Montreal General Hospital pregnant nuclear energy workers do not work with nuclear substances (Colbalt teletherapy or brachytherapy), for which emergency situations may present the possibility of increased dose. An emergency situatioin with a linac, on the otherhand, should not present the risk of increased dose as a linac can simply be turned off in an emergency.
- A person can exceed all dose limits when acting voluntarily to save or protect human life.
- When a dose limit is exceeded specific actions are required. These include:
  - Immediately notify the person and the CNSC
  - Require the person to leave any work that may add to the dose
  - Conduct an investigation to determine the magnitude of the dose and to establish the cause of the exposure
  - Identify and take any action required to prevent a similar incident from occuring again
  - Report to the CNSC within 21 days the results or progress of the investigation
- The person who has exceeded the dose limit may not return to work until authorized by the CNSC. Authorization may include restrictions.

Organ/ Tissue	Person	Period	Dose (mSv)
	Effective Dose Limits		
Whole body	Nuclear energy worker, including a pregnant nuclear energy worker	(a) One-year dosimetry period*	50
		(b) Five-year dosimetry period	100
	Pregnant nuclear energy worker	Balance of the pregnancy	4
	A person who is not a nuclear energy worker	One calendar year**	1
	Equivalent Dose Limits		
Lens of an eye	(a) Nuclear energy worker	One-year dosimetry period	150
	(b) Any other person	One calendar year	15
Skin	(a) Nuclear energy worker	One-year dosimetry period	500
	(b) Any other person	One calendar year	50
Hands and feet	(a) Nuclear energy worker	One-year dosimetry period	500
	(b) Any other person	One calendar year	50

Table 6.1: CNSC effective and equivalent dose limits.
* One-year dosimetry period is defined as beginning on January 1st and ending on December 31st.
** One calendar year is defined simply as a period of 12 months.

- The requirement to label all containers or devices containing nuclear substances, with
  - the radiation warning sign (trefoil), and the words "RAYONNEMENT—DANGER—RADIATION", and
  - the name, quantity, date of measurement and form of the nuclear substance in the container or device
- The requirement to post the radiation warning sign, and the words "RAYONNEMENT—DANGER—RADIATION" outside rooms that
  - contain greater than 100 EQ of a radioactive substance
  - present a reasonable probability that a person entering the room will be exposed to a dose rate greater than 25 μSv/hr.

Nuclear Substances and Radiation Devices Regulations

The nuclear substances and radiation devices regulations regulate the amounts of radioactive substances that may be used in Canada without a license (<1 EQ), not without a radiation warning sign (>100 EQ) and not without special CNSC

permission (>10 000 EQ). Special exemptions are made for smoke detectors, tritium safety signs, and check sources (such as those used with survey meters).

These regulations also mandate leak tests, to be performed on sealed sources containing 50 MBq or more. A leak test should be performed immediately before using the sealed source, if it has been in storage for the last 12 months, or every 24 months if the source is in long-term storage. Furthermore, leak tests should be performed immediately if a source is damaged. Sealed sources contained within radiation devices (for example, a Cobalt-60 teletherapy device) must be leak-tested every 12 months.

Class I Nuclear Facilities Regulations

Class I facilities are sub-categorized as class IA and Class IB facilities. Class IA facilities refer to facilities that contain nuclear reactors. Class IB facilities are facilities that process uranium, thorium or plutonium or they are facilities that are not covered under the definition of Class II facilities, as defined in the Class II regulations.

Interesting highlights of the class I nuclear facilities regulations include:

- The definition of an "exclusion zone" as a parcel of land within or surrounding a nuclear facility on which there is no permanent dwelling and over which a licensee has the legal authority to exercise control.
- The requirement that the licensee undertake measures to prevent acts of sabotage or attempted sabotage (ie nuclear terrorism).

Class II Nuclear Facilities and Prescribed Equipment Regulations

Class II nuclear facilities and prescribed equipment regulations are of most interest to medical physicists. A class II nuclear facility is defined as a facility

that includes Class II prescribed equipment. Class II prescribed equipment is defined to cover many types of radiation-emitting devices including

- Irradiators with more than 10^{15} Bq of nuclear substance
- Irradiators that require external shielding and which can deliver dose rates of more than 1 cGy/min at 1 m.
- A radioactive source teletherapy machine.
- Particle accelerators up to 50 MeV if the particles have a mass less than or equal to 4 atomic mass units.
- Particle accelerators up to 15 MeV if the particles have a mass greater than 4 atomic mass units.
- A brachytherapy remote afterloader.

An exact list of class II prescribed equipment certified by the CNSC for use in Canada, as of September 2010, can be found on the CNSC's website.

After defining the types of equipment that fall under the class II category, some details regarding license applications for class II facilities are outlined. Class II facilities are required to apply for a license to construct before construction of the facility begins, a license to operate before the equipment can be used at the facility and a license to decommission before the equipment can be un-installed. The equipment itself must also be certified (ie approved by the CNSC) before it may be used.

The class II nuclear facilities and prescribed equipment regulations provide detailed requirements pertaining to radiation protection at class II facilities. The radiation protection requirements mandate the following for rooms containing class II prescribed equipment:

- A door interlock that prevents the equipment from being used while the door is open.

- Last-person out safety check buttons that must be pressed, in the correct sequence, inside the room and then outside the room, in order to operate the equipment.
- A mechanism to ensure that no person can be locked inside the room.
- A system to allow the operator to view the interior of the room during operation of the equipment.
- A visible display (ie warning light) outside of a room that indicates the irradiation state of the equipment contained within it.
- An area radiation monitoring system if the room contains prescribed equipment other than a particle accelerator.
  - the reason for area monitoring in rooms not containing particle accelerators is to provide a visual warning, independent of the device, that the source is in the unshielded position. Particle accelerators, since they require electricity to function, can be reliably stopped by simply removing the power supply.
- Emergency stop buttons at various locations inside and outside the room.
- Posting of emergency contact details at the entrance to the facility.

Radiation Safety Officers (RSOs) are mandated by the class II nuclear facilities and prescribed equipment regulations. An RSO must be certified by the CNSC after having undergone a CNSC examination.

Some special regulations pertaining to brachytherapy remote afterloaders are specified:

- All patients who undergo brachytherapy treatments must be surveyed with a survey meter before they are released.
- A room containing a brachytherapy remote afterloader should contain a shielded storage container and remote handling tools to recover radioactive sources in an emergency.

Radiation survey meters must be made available to each worker in a class II nuclear facility. The survey meter must meet the following requirements:

1. Have been calibrated within the last 12 months.
2. Must be cable of measuring the x-ray, gamma-ray and, if applicable, neutron radiation from the equipment.
3. Must have a battery power-level indicator.

Class II prescribed equipment cannot be used on a person (ie patient) unless directed by a qualified medical practitioner.

Various records regarding the details of the class II prescribed equipment and its usage must be retained. Such records include the daily output of the machine and details of worker training. All records should be kept for three years after the expiry date of the class II license.

6.0.3 Radiation Emitting Devices

The import, export, leasing, buying and selling of radiation-emitting devices for certain medical, industrial and consumer applications are controlled by the Radiation Emitting Devices (RED) Act of 1985 and associated regulations drawn up by Health Canada. The RED regulations to not apply to devices that are designed for the production of nuclear energy (such devices are regulated by the CNSC under the NSCA). They also do not regulate the use of radiation-emitting devices, once purchased. They do, however, cover the purchase of x-ray tubes (for diagnostic radiology), x-ray machines and linear accelerators (for radiation therapy) as well as UV tanning equipment. Manufacturers intending to sell radiation emitting devices within Canada must first obtain Health Canada approval in order to do so.

6.0.4 Transport Regulations for Radioactive Material and Devices

The packaging and transport of radioactive substances within Canada is regulated by the CNSC in cooperation with Transport Canada. The transportation of dangerous goods, in general, is regulated by the Transportation of Dangerous Goods (TDG) Act of 1992 and associated regulations, while the regulations governing the transport of radioactive material, in particular, is codified in the CNSC's "Packaging and Transport of Nuclear Substances Regulations".

The TDG Act stipulates certain requirements for the labelling and transport of a package containing dangerous goods. Since dangerous goods are frequently shipped internationally, Canada adheres to the recommendations drawn up by the United Nations Sub-Committee of Experts on the Transport of Dangerous Goods . Specific international recommendations for the transport of radioactive substances are found in the IAEA Safety Standards reports pertaining to transportation.

UN recommendations specify a coloured diamond-shaped label for the packaging of dangerous goods and nine classes of goods. The nine classes are listed in table 6.2. Radioactive materials fall under class 7. Four categories of class 7 goods are defined, each with a different placard. Categories I, II and III include radioactive materials with increasing activities and transport indices. The fourth category is for fissile material. Class 7 radioactive goods must display a label that specifies the following information:

- the name(s) or symbol(s) of the nuclear material,
- the activity of the nuclear material, and
- the transport index of the nuclear material.

The category into which the radioactive goods fall, and hence the labelling of the package used to transport them, is determined by the dose rate at the external surface of the package and the transport index of the package (ie dose

Table 6.2: The nine classes of dangerous goods defined by the UN. The descriptions of each class and the appropriate placards are provided on the Transport Canada website.

Class	Goods
1	Explosives
2	Gases
3	Flammable Liquids
4	Flammable Solids
5	Oxidizing Substances and Organic Peroxides
6	Toxic Substances and Infectious Substances
7	Radioactive Materials
8	Corrosives
9	Miscellaneous Products, Substances or Organisms

Table 6.3: The signs used for the packaging of Class 7 radioactive materials, according to measured dose rate at the external surface and the transport index.

Group Number	Sign	External Surface		Maximum Transport Index
		Lower Limit (mrem/hr)	Upper Limit (mrem/hr)	
I	White	0	0.5	n/a
II	Yellow	0.5	50	1
III	Yellow	50	200	10

rate at 1 m from the external surface). Table 6.3 provides the breakdown of labels and corresponding external dose rates and transport indices. Figure 6.1 presents the four class 7 radioactive material signs.

CANUTEC

In the event of an emergency involving the transport of dangerous goods, a Transport Canada phone helpline, known as CANUTEC (Canadian Transport Emergency Centre) is available to provide advice. The CANUTEC phone number is 613-996-6666 or *666 on a cellular phone.

6.1 Quebec Regulations Pertaining to Radiation Safety in Medicine

Quebec regulations pertaining to the use of radiation in private medicine (ie in private dental and radiology clinics) are found in the Quebec Civil code in section R.S.Q. c. L-0.2; “An Act respecting medical laboratories, organ, tissue, gamete and embryo conservation, and the disposal of human bodies”.

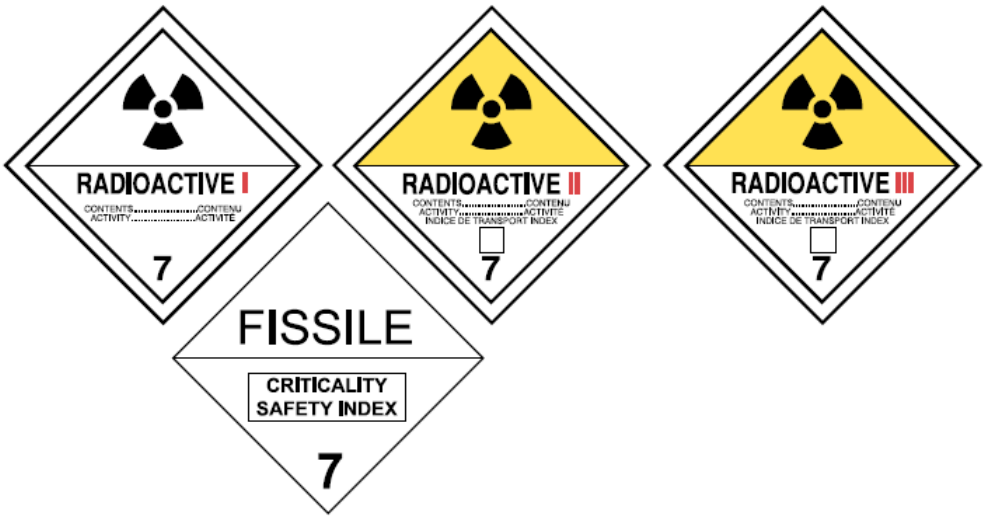


Figure 6.1: The four placards used in the packaging and transport of Class 7 radioactive materials.

This act replaces the previous "Public Health Protection Act". Quebec regulations dealing with radiation in hospitals (which are public institutions) are covered by the "Organization and Management of Institutions" regulation (c. S-5, r.3.01), which specifies that hospitals "shall put into practice methods of controlling the use of appliances, such as those prescribed by the Regulation respecting the application of the Public Health Protection Act".

CHAPTER 7
Radiation Protection In Practice

At this stage we should have a good understanding of what radiation is, how it can affect our health and how we can detect and quantify it. We have seen that there are legal restrictions on the use and handling of radioactive substances and radiation emitting devices - imposed by the federal and provincial governments for our safety. We have also seen that there are scientific and regulatory organizations available to guide us in our radiation protection efforts. With radiation protection theory examined, it is now time to focus on how we can apply radiation protection principles to practical situations where radiation is encountered.

According to the ICRP (reports 103, 60, etc), the three principles of radiation protection - justification, optimization and application of dose limits - apply to all situations in which radiation is used. Similarly the three tenets of physical radiation protection - distance, time, and shielding - apply to all instances of actual radiation exposure. In this chapter we will use these two triads of radiation protection together with practical recommendations from various guidance documents and federal and provincial regulations to enumerate the main components of a practical radiation protection program.

7.1 Context

In approaching the task of implementing a radiation protection program it is important to first put the planned radiation exposure situation into context. Three questions help contextualize the exposure situation:

1. What type of exposure situation is expected?

2. What category of individuals will or might be exposed?
3. Which regulations apply?

7.1.1 Type of Exposure Situation

According to the ICRP, three main types of exposure situation can be envisaged; planned, emergency and existing exposure situations. Additionally, the ICRP recommend that medical exposure, although it technically falls under planned exposure, should be dealt with separately owing to the different radiation protection considerations involved. The following definitions are taken directly from ICRP report 103:

Planned exposure situations are situations involving the deliberate introduction and operation of sources. Planned exposure situations may give rise both to exposures that are anticipated to occur (normal exposures) and to exposures that are not anticipated to occur (potential exposures).

Emergency exposure situations are situations that may occur during the operation of a planned situation, or from a malicious act, or from any other unexpected situation, and require urgent action in order to avoid or reduce undesirable consequences.

Existing exposure situations are exposure situations that already exist when a decision on control has to be taken, including prolonged exposure situations after emergencies.

Medical exposure situations are exposure situations that are planned in advance for the purpose of diagnosis, intervention, therapy or biomedical research.

7.1.2 Category of Exposed Individuals

The ICRP distinguish between three categories of exposed individuals—occupationally exposed workers, members of the public and patients exposed for medical reasons. While recognizing that workers might also be exposed to ionizing radiation as members of the public or as patients, the commission recommends against attempting to combine, for regulatory purposes, exposures from each category.

Workers

A worker is defined by the ICRP as

> any person who is employed, whether full time, part time, or temporarily, by an employer and who has recognized rights and duties in relation to occupational radiological protection.

Although no distinction is made between the sexes, special consideration is recommended for female workers who are pregnant or nursing. Additional radiation protection controls are must be considered in order to protect the embryo/fetus.

Members of the Public

In ICRP report 103, a member of the public is defined as

> as any individual who receives an exposure that is neither occupational nor medical.

In considering measures to protect members of the public the ICRP recommend use of the concept of the Representative Person, who is representative of the group of individuals in the population who are most exposed to background radiation sources - both man-made and natural.

Patients

The ICRP define a patient as

> an individual who receives an exposure associated with a diagnostic, interventional, or therapeutic procedure.

Dose limits are not recommended for patients but the principles of justification and optimization of dose still apply, see section 7.2 below.

7.1.3 Regulatory Jurisdiction

As outlined in chapter 6, in Canada all matters pertaining to the control of exposure due to nuclear substances (ie the control of practices involving radioactive materials and radiation devices that may induce activation) are regulated on behalf the federal government by the CNSC. Jurisdiction for the control of exposure due to radiation-emitting devices that do not induce activation (eg medical and dental x-ray tubes, CT scanners and fluoroscopes) are regulated by provincial governments. Any practice involving radiation exposure and associated protection programs must adhere to the appropriate federal and/or provincial regulations, as appropriate.

7.2 Applying the Three Main Principles of Radiation Protection

The three main principles of radiation protection as promulgated and refined in numerous ICRP publications apply to all types of exposure situation and to all categories of exposed individuals. Actual implementation of the principles will depend on the exposure context (ie, exposure type and category of exposed individuals).

7.2.1 The Principle of Justification

The principle of justification plays a pivotal role in the establishment of any radiation protection program. Is the proposed exposure activity justified? Is

the radiation really needed? Is there another, safer, way of performing the activity without the use of radiation? In their most recent recommendations (ICRP Report 103) the ICRP present the principle of justification as follows:

> **The principle of justification:** Any decision that alters the radiation exposure situation should do more good than harm.

It is interesting that the ICRP use the word *alter* rather than increase. Essentially they recognize that the consequences of a justification decision are not confined solely to the use of the radiation. Rather, the activity as a whole, of which the use of radiation may be just one part, needs to be considered comprehensively.

Application of the principle of justification depends on whether the activity involving radiation exposure is planned in advance, is unplanned or is for medical purposes.

Planned Exposure. Introduction of a planned activity (ie a new activity) should not proceed unless it offers sufficient net positive benefit to the individuals exposed or to society as a whole to offset the radiation detriment it causes. The introduction of a planned exposure activity will only increase the dose (and hence detriment) to the exposed population. As such, radiation protection measures should be planned in advance and applied directly to the source of the radiation from the outset.

Unplanned exposure. Action should not be taken to change an unplanned exposure situation (ie an emergency or an existing exposure situation) unless the action causes net positive benefit. It is recognized here that any actions taken to avert future exposure may also have disadvantages in terms of the resultant balance of good versus harm to society.

Medical exposure. The aim of medical exposure is to do more good than harm to the patient (or to society as a whole through the exposure of volunteers for biomedical research). As such, medical exposure should be intrinsically justified, with due regard for the well-being of the medical staff involved.

Unjustified Exposure. Some exposure situations are never justified. Three categories of unjustified exposure are identified by the ICRP:

1. Exposure due to the deliberate addition of radioactive substances to food and personal objects such as toys, clothing and jewellery.
2. Radiological exposure for non-medical purposes. Examples might include examinations for employment, health insurance and legal purposes. Examinations for important security purposes may be justified in certain circumstances. Owing to increased airport security following the terrorist attacks of September 11 2001, exposure due to airport screening is of interest to the ICRP. Presently a draft report on the subject of radiation protection in security screening is presented for consultation on the ICRP's website (www.icrp.org).
3. Medical screening of asymptomatic population groups. Medical screening to search for disease is considered unjustified unless the expected advantages to the individuals examined and to society as a whole are sufficient to compensate for the radiation detriment involved. Of interest here might be considerations such as the frequency at which a woman should receive a mammogram or the acceptability of whole body CT screening for desiring individuals who are willing to pay for it.

Decision Making and Licensing

Judgments involving the principle of justification for planned exposure (excluding medical exposure) and unplanned exposure situations are seldom made by individuals. Rather, the responsibility for such judgments falls on governments or governmental organizations that comprehensively examine the activities involved and their benefits and drawbacks for society. In practice, justification is manifested by licensing procedures through which the regulatory authority licenses the user of the radiation to carry out the activity that is justified. Justification decisions regarding medical exposure, on the other hand, are made by the medical profession generally and by individual medical practitioners on a patient-by-patient basis.

Cost-Benefit Analysis

In applying the principle of justification a cost-benefit analysis may be used to aid the decision maker. Cost-benefit analyses are the livelihood of actuarial scientists and insurance companies. Using the ICRP's principle of justification, the goal of a cost-benefit analysis for a proposed radiation exposure situation is to determine if the benefit of the proposed activity is greater than the expected harm caused by it (i.e. is the benefit greater that the expected detriment to the exposed population?). In general, a cost-benefit analysis may be summarized by the equation

$$B = V - C$$

where

- B is the net benefit to be gained from the proposed action
- V is the gross benefit
- C represents all the costs related to the proposed action

In the case of a radiation exposure situation, V would be the clear benefit derived from the radiation usage (for example electricity from a nuclear power plant or images from a diagnostic radiology procedure). C would represent the sum of the production costs P (for example the cost of constructing a power plant or radiology room and paying salaries) and the costs of both the level of radiation protection X that is implemented and the cost of the detriment Y resulting from residual radiation exposure beyond the level of protection.

Accordingly, for a given radiation exposure situation an appropriate cost-benefit analysis equation might be

$$B = V - [P + X + Y] \tag{7.1}$$

A cost-benefit analysis of this type is described in ICRP report 37.

7.2.2 The Principle of Optimization

The principle of optimization comes into play if and when an exposure situation has been justified. It is the principle of most practical interest to the health physicist. The principle of optimization of protection is defined by the ICRP as follows:

> **The principle of optimization:** The likelihood of incurring exposure, the number of people exposed, and the magnitude of their individual doses should all be kept as low as reasonably achievable, taking into account economic and societal factors.

It is clear from the wording that the principle of optimization is in essence the ALARA principle. In theory the cost-benefit analysis outlined above can be extended to perform an optimization analysis. The gross benefit V and the production costs P are fixed and presumably quantifiable for any radiation exposure situation. The cost of radiation protection X and the cost of the

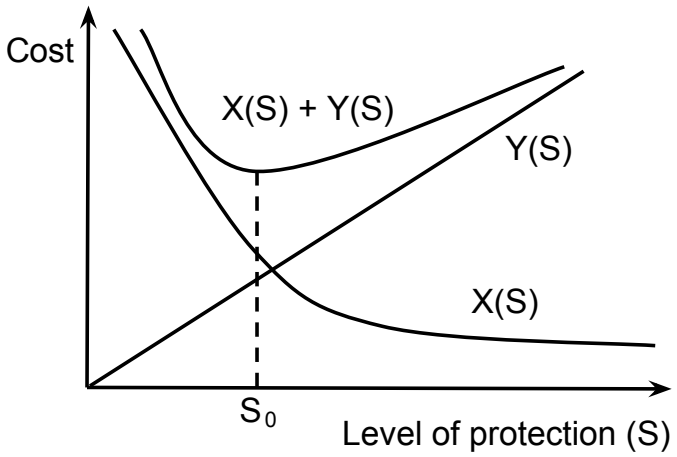


Figure 7.1: A process of optimization for radiation protection can be considered as a search for the level of protection S_0 that results in the minimum of the sum of the radiation protection cost X(S) and the cost of the detriment Y(S). S is the collective effective dose due to the residual radiation exposure to the population and is one quantity that may be used to represent the level of protection. Figure based on the IAEA basic safety standards IAEA (1982).

detriment due to residual exposure Y are not fixed but depend on the level of radiation protection that is implemented. Thus, the optimum level of radiation protection, represented, for example, by the collective effective dose received by the exposed population, would be the level for which the sum X(S) + Y(S) is minimized, as shown in Figure 7.1. This type of radiation protection optimization is discussed in the IAEA's Basic Safety Standards for Radiation Protection (BSS) (IAEA, 1982).

The principle of optimization applies to medical exposure situations. The dose to the patient should be kept as low as is required to produce the desired outcome. For diagnostic and interventional exposures, the dose delivered to the patient should be just sufficient to provide the necessary levels of constrast, noise and resolution. The image aperature should be limited to the anatomical region of interest and care should be taken to ensure that the first image is taken correctly, such that repeat imaging is unnecessary. Fluoroscopic

imagaing procedures should always make use of a "last image hold" function. Therapeutic procedures should be optimized such that the prescribed dose is delivered to the target volume with minimum dose to surrounding healthy tissue. Furthermore, the amount of healthy tissue in the target volume should be minimized through appropriate image guidance techniques.

7.2.3 The Principle of Application of Dose Limits

The principle of application of dose limits applies to all planned non-medical radiation exposure that has been justified. The principle of application of dose limits is defined by the ICRP as follows:

> **The Principle of Application of Dose Limits:** The total dose to any individual from regulated sources in planned exposure situations other than medical exposure of patients should not exceed the appropriate limits specified by the Commission.

The ICRP recommend that dose limits may be relaxed in emergency exposure situations where informed individuals are engaged in voluntary life-saving actions. However, responders undertaking recovery and restoration operations at the later phase of an emergency should be considered as occupationally exposed.

The ICRP recommend that the level of protection afforded to the embryo/fetus should be broadly similar to that of a member of the public (ie 1 mSv/year). Furthermore, it is recommended that, although pregnant workers need not be removed completely from work practices involving radiation, their working conditions should be such that the probability of accidental exposure is very low. In practice this might mean removing a pregnant worker from a situation in which an emergency exposure situation may arise.

Generally, it is the relevant regulatory authority that sets the dose limits for radiation exposure and these may or may not be consistent with the limits recommended by the ICRP. In Canada, the regulatory limits promulgated by the CNSC are broadly consistent with those of the ICRP, except in the case of pregnant nuclear energy workers, for which the Canadian limit of 4 mSv for the balance of pregnancy is more liberal.

In Quebec, maximum permissible dose limits for x-ray exposure are prescribed schedule 8 of the *Regulation respecting the application of the Act respecting medical laboratories, organ and tissue conservation and the disposal of human bodies*. Again, the Quebec limits are broadly similar but not exactly the same as the CNSC limits and the ICRP recommendations.

7.3 Implementing a Radiation Protection Program

Once a radiation exposure situation has been justified (eg a license obtained from the relevant regulatory authority), the principles of optimization of protection and application of dose limits come into play. A solid radiation protection program is required to ensure that all exposures are kept ALARA, social and economic factors accounted for, and that the regulatory dose limits are never reached.

The framework for a radiation protection program should include the following four basic components:

1. A solid management structure
2. A radiation protection committee
3. A radiation safety officer, and
4. A radiation safety manual outlining policies and procedures

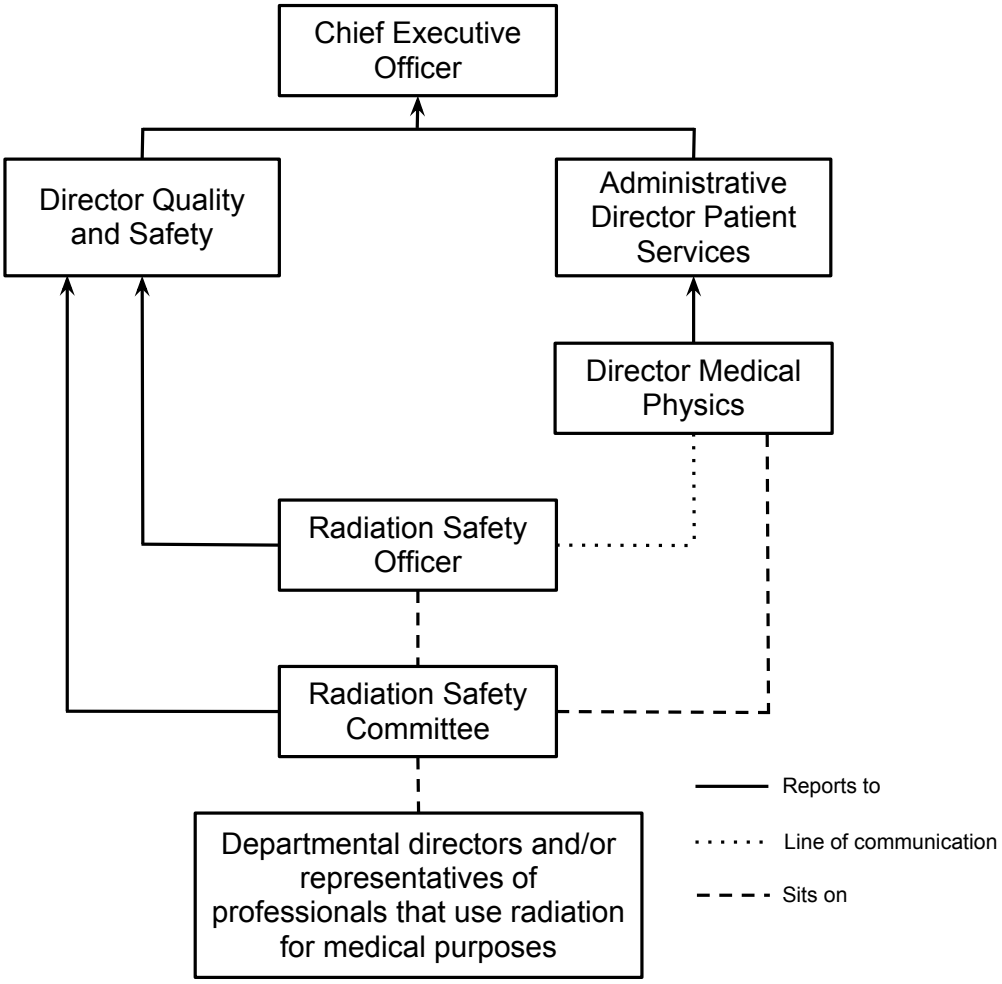


Figure 7.2: Organigram showing the organization of communication and reporting for a generic hospital radiation safety program.

Management Structure

Just as is the case for other health and safety programs, a radiation protection program cannot exist unless it is underpinned by a solid management structure. Roles and responsibilities with regard to radiation protection should be clearly delineated. Organigrams are often used for this purpose. An organigram is a chart that illustrates the relationships between different members of an organization. Figure 7.2 presents a generic radiation safety organigram that might be used in a hospital.

A radiation safety officer with competency in radiation protection should be appointed. Indeed, an RSO is usually mandated by the applicable regulations. A multi-professional radiation safety committee should also be formed and entrusted with the tasks of monitoring the radiation protection program and reaching consensus on important radiation protection decisions. The roles of the RSO and RPC are discussed in greater detail below.

Commitment to the principle of ALARA by senior management is vital in order for a radiation protection program to succeed. Often a statement of support is written into the radiation safety manual. The commitment should be backed up with practical support in terms of the resources needed (man-power, time and money) to implement an optimized radiation protection program.

Radiation Protection Committee

The role of the radiation protection committee is to oversee the radiation protection program, to ensure compliance with the principle of ALARA and with federal and provincial regulations. The committee is responsible for reaching consensus on important decisions regarding radiation protection and for procuring, from senior management, the resources necessary to maintain the radiation protection program.

The radiation protection committee is the focal point for the radiation protection program and as such it should be multi-professional in composition. It should include the RSO and representatives of all the radiation users within the organization. For example, in a hospital the radiation protection committee might include representatives of physicians, nurses, therapists, physicists, senior management and the department of quality and safety. The committee should meet regularly, for example once per quarter, to review the radiation safety program.

Radiation Safety Officer

The RSO is responsible for the implementation, coordination, and day-to-day oversight of the radiation protection program (AAPM Task Group 160, 2010). The RSO is not normally responsible for the direct supervision of radiation users in their daily work but he/she should have the authority to enforce policies and procedures dealing with radiation safety and regulatory compliance. RSO responsibilities include the following:

- Sit on the radiation protection committee
- Oversee the operational aspects of the radiation protection program and delegate tasks as appropriate
- Ensure compliance with applicable regulations and license requirements
- Help identify and resolve radiation protection problems
- Stop unsafe practises
- Notify management and the regulatory authorities of radiation safety problems and corrective actions, as appropriate
- Liaise with the regulatory authorities (ie paperwork)
- Act as the contact person for all matters related to radiation safety
- Perform leak checks as mandated in the CNSC regulations and submit compliance reports as appropriate
- Coordinate education and training of staff for matters pertaining to radiation safety
- Maintain a calibrated working inventory of radiation safety equipment
- Designate nuclear energy workers
- Edit/maintain the radiation safety manual

The the presence of an RSO may be mandated by the applicable regulations. For example, the CNSC Class II regulations mandate an RSO who is

certified by the CNSC after undergoing a CNSC certification exam. According to the regulations, the RSO must have a designated replacement (DR), essentially a second person in the institution who is qualified to perform the duties of the RSO in his/her absence. The RSO or DR must be present at the licensed Class II facility for at least a portion of each day and must be contactable off-hours when the class II equipment is operated by "qualified persons". Backup RSOs are also mandated but need not be certified by the CNSC. Backup RSOs may perform some of the duties of the RSO (such as dealing with shipping/receiving, emergencies, pregnant nuclear energy worker designation and minor contact with the CNSC) when neither the RSO nor the DR are on site.

Radiation Safety Manual

An updated radiation safety manual is a very useful reference within an organization regarding radiation protection policies and procedures. It should cover all aspects of radiation safety policy within the organization and copies should be accessible to all members of staff.

Components of a Radiation Safety Program

The following is a list of items and activities (some of which have been mentioned already) that comprise a typical radiation safety program:

- ALARA policy
- Radiation safety officer
- Radiation safety committee
- Radiation safety manual
- Education/training for staff
- Shipping/receiving of radioactive substances
- Waste disposal

- Dose limits including limits for pregnant nuclear energy workers
- Nuclear energy worker designation
- Dose monitoring (personnel and area)
- Emergency planning
- Paperwork (e.g., regulatory communications, records)
- Signage
- Inventory and calibration of radiation protection equipment
- Facility design and modification
- Incident reporting
- Emergency planning/response
- Leak testing

CHAPTER 8
Radiation in Healthcare

The discovery of x-rays at the end of 1895 represents one of the finest examples of translational science - Roentgen's demonstration that he could image human anatomy (namely his wifes hand) went from laboratory experiment to medical practise within weeks of his reporting it. In fact, the first Canadian x-ray image was taken in early 1896 at the Montreal General Hospital by McGill physics professor John Cox just a few months after Roentgen's publication. Today the use of ionizing radiation is firmly established in diagnostic and therapeutic medicine. Currently approximately 4 billion medical procedures with ionizing radiation are performed annually worldwide, making medical radiation the largest, and still growing, artificial contribution to background radiation.

Radiation has three main applications in medicine:

- Diagnosis
- Intervention, and
- Therapy

Additionally radiation is used in healthcare for biomedical research, for the sterilization of blood and medical equipment and in the production of radionuclides for nuclear medicine.

In this chapter we will briefly examine each of the above modalities and outline associated radiation protection considerations.

Please refer to the notes presented in class.

CHAPTER 9
Radiation Shielding for Medical Linear Accelerators

Internationally accepted guidelines for the design of structural shielding for radiation therapy installations are laid out in the National Council for Radiation Protection report number 151 (NCRP, 2005) and in the Institute of Physics and Engineering in Medicine (IPEM) report number 75 (IPEM, 1997), herein referred to as just NCRP 151 and IPEM 75, respectively. These reports detail the calculations involved in determining the barrier thickness needed to shield an individual from a source of radiation such that his/her effective doses are kept well below the appropriate maximum permissible values. They also provide general recommendations for the design of shielding around radiation therapy installations. A third report, the IAEA Safety Reports Series (SRS) No. 47. (IAEA, 2006), draws upon both NCRP 151 and IPEM 75 and is available free of charge on the IAEA website[1] .

This chapter describes the calculation methodology of the NCRP report, currently the standard shielding design document used for radiation therapy facilities in North America. This report, along with the NCRP report number 49, was used in the design calculations for the Radiation Oncology department at the Montreal General Hospital.

9.1 Equivalent and Effective Doses

The NCRP 151 authors use equivalent dose H as the single dose quantity of interest in all shielding calculations. They do so, despite the fact that

[1] http://www-pub.iaea.org/MTCD/publications/PDF/Pub1223_web.pdf

the ICRP and NCRP dose limit recommendations and the legal maxima prescribed by regulatory bodies, including the CNSC, are for both equivalent dose and effective dose. The reason to use equivalent dose alone is a matter of practicality—it is simply unfeasible to estimate effective dose values without exact prior knowledge of quantities such as the position, size and posture of the person who is exposed.

9.2 Production of Therapeutic Radiation Beams

Megavoltage therapeutic photon and electron beams are typically produced using linear accelerators. To produce an X-ray beam, a linear accelerator accelerates electrons toward a target whereupon bremsstrahlung X-ray photons and heat are produced. To produce an electron beam, the target is removed. As shown in Figure 9.1, there are several major components to a therapeutic linear accelerator: the gantry, the gantry stand and support, the head, the patient couch and the control console.

Depending on the type of linear accelerator involved, the radiation beam may be formed in the gantry stand or within the gantry itself (as is the case in Figure 9.1). Electrons are produced by means of thermionic emission and acceleration off a heated cathode within a device known as an *electron gun*. From the electron gun, the electrons drift into an accelerating RF waveguide. Within the waveguide they are accelerated to MeV energies as a result of energy transfer to them from the high-power RF field. The beam transport system controls delivery of the energetic pulsed electron beam from the waveguide to the target (for an X-ray beam) or to a scattering foil (for an electron beam). The purpose of the scattering foil is to spread out the otherwise narrow electron beam. Before reaching the patient, the beam is shaped within the head of the unit using collimating jaws made of high-Z material or using a

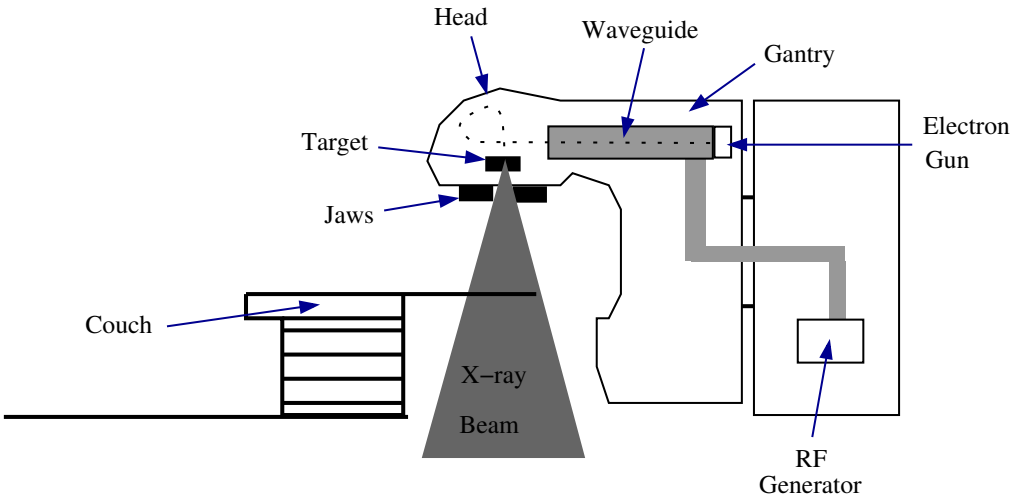


Figure 9.1: The main components of a linear accelerator used to produce megavoltage therapeutic photon and electron beams.

computer-controlled multileaf collimator (MLC). Additional details regarding the geometry of radiation therapy treatment rooms is provided in section 9.3.

9.3 Treatment Room Geometry and Sources of Radiation

NCRP 151 considers megavoltage photon as well as electron beams used for radiation therapy. The applicable energy range varies from 1.25 MV to 24 MV. At these energies, linacs or cobalt-60 teletherapy units are employed and, as discussed above, the radiation source is usually incorporated into a gantry that rotates in a single plane around the patient. The horizontal axis of rotation is referred to as the isocenter axis. The isocenter itself is the volume defined by the intersection of the isocenter axis and the axes of rotation of both the patient couch and the collimator of the radiation unit, and is typically close to the center of the treatment room. Figure 9.2 presents schematics of a typical radiation therapy treatment room, showing the position of the isocenter and the layout of the walls, ceiling and door.

Two types of radiation reach the walls of the room and must be accounted for in shielding design calculations: (1) primary beam (either attenuated through the patient or unattenuated) and (b) secondary radiation arising

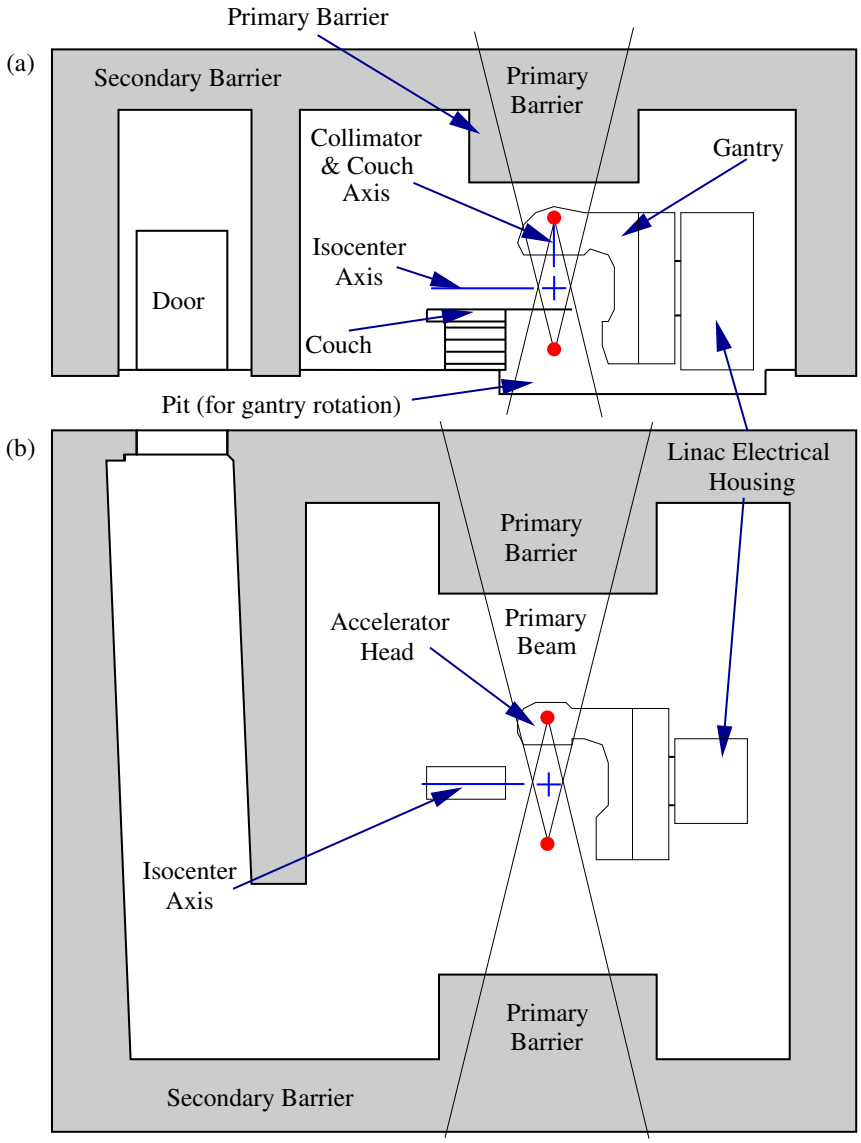


Figure 9.2: Schematics showing the geometry of a typical radiation therapy treatment room. (a) Front elevation view. (b) Plan view. The location of the isocenter is shown by a blue cross and the source positions for beam directions perpendicular to the viewing angle are marked by red dots.

from leakage through the shielded head of the radiation generator and from scattered radiation produced by the interaction of the primary beam within the patient. Figure 9.3 is an illustration of the radiation beams of interest, together with the primary and secondary barriers used to shield them. Since the energy of the primary and secondary beams differ significantly, they must be considered independently and require separate shielding design calculations. The primary beam is a true beam in the sense that it is collimated in the direction of the isocenter, and thus "beamed" onto the wall, whereas the secondary radiation may travel in all directions and is usually assumed to originate at the isocenter[2] .

At high energies, above ∼10 MV, two additional sources of secondary radiation must be considered; photoneutrons produced by photonuclear interactions within the machine head and within the patient, and subsequent gamma rays produced by neutron-capture interactions. Since neutrons scatter freely, they, and the capture gamma rays they produce, are also considered as multidirectional beams of secondary radiation emanating from the isocenter.

In addition to appropriate primary and secondary barriers, a treatment room must incorporate a shielded door for access, as shown in Figure 9.3. At low energies the door design is straightforward since only photons are involved. Above 10 MV, however, the presence of photoneutrons complicates matters considerably. A direct shielded neutron door (i.e., a door directly exposed to the secondary neutron beam) is often so large and heavy that it is impractical.

[2] Scatter radiation is actually produced from all points at which the primary beam encounters matter. Nevertheless, the assumption that it originates, on average, at the isocenter is valid when all gantry angles used in patient treatments are considered. It is also assumed that scatter by air molecules is negligible

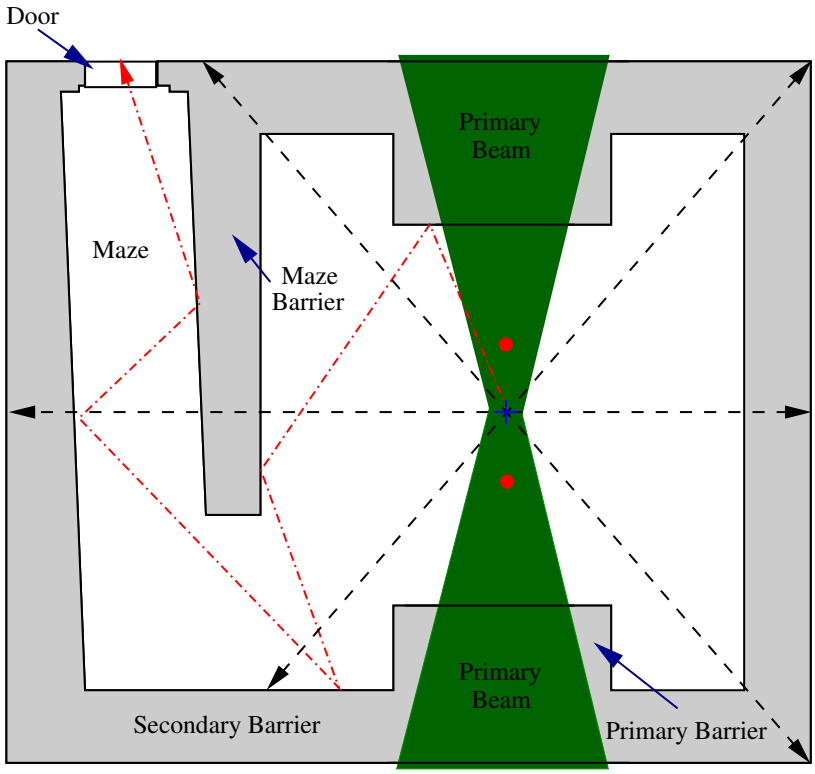


Figure 9.3: The primary (shaded green) and secondary (lines) radiation beams produced inside a radiation therapy room and the barriers used to provide shielding against them. Several secondary beams are shown—the black dashed line represents leakage and scatter, the dot-dashed line represents photoneutrons. As described in the text, all appear to emanate from the isocenter when all gantry angles are accounted for. The location of the isocenter is shown by a blue cross and the source locations for 180° gantry rotations are marked by red dots.

Additional secondary barrier shielding to attenuate the secondary radiation before it reaches the door is required. A "maze" wall, forming a corridor from the door to the far wall of the room, is the solution. The treatment room depicted in Figures 9.2 and 9.3 incorporates a maze.

9.4 Shielding Materials

The shielding properties of a number of materials are detailed in chapter 4 of NCRP 151. Table 9.1 provides a summary. A comparison of materials used for photon shielding in megavoltage radiation therapy is quite straightforward since only a density comparison is required. This is due to the dominance of the Compton effect at megavoltage energies (see Figure 2.6) and the Z-independence of the Compton effect. However, with regard to shielding radiation from linear accelerators that accelerate electrons to energies above 10 MV, the ability of an attenuating material to absorb or moderate neutrons is an important additional consideration. High-Z materials, such as lead and steel are nearly transparent to fast neutrons, although they do moderate their energies. Materials with high hydrogen content (e.g., concrete) or borated materials (such as borated-polyethylene) are the best neutron absorbers.

9.5 Overview of Shielding Calculations

The purpose of any radiation shielding calculation is to determine the thickness of a barrier needed to reduce the radiation dose received at a point of interest (POI) at one side of the barrier, due to a radiation source at the other side, to a desired level. In NCRP 151, the desired radiation level is called the "design goal" P. Its value is equal to the regulatory dose limit modulated by an appropriate ALARA factor (see section 4.5.2). Figure 9.4 graphically depicts the geometry, distances and quantities involved in a shielding calculation.

Table 9.1: Summary of the properties of the shielding materials encountered in NCRP 151. Primary barrier thicknesses equivalent to 2.5 m of ordinary concrete are provided.

Material	Density ($g\ cm^{-3}$)	Hydrogen Content $\times 10^{22}$ atoms cm^{-3}	Primary Barrier Thickness (m)	Relative Cost	Advantages	Disadvantages
Ordinary Concrete	2.35	0.8 to 2.4	2.5	$$	Easy to pour/configure, relatively inexpensive, easily available, good X-ray shielding, good neutron shielding, structurally strong	Large footprint in terms of barrier thickness
HD Concrete	> 2.35	0.8 to 2.4	1.4	$$$$	Small footprint compared to ordinary concrete	Handling difficulties. Primary barrier adequate for photons not necessarily adequate for neutrons
Lead	11.35	None	0.5	$$$	Thin barriers	Lead is malleable and needs to be supported against its own weight. Toxic. Nearly transparent to fast neutrons
Steel	7.8	None	0.8	$$	Relatively thin barriers, strong, less expensive than lead	Nearly transparent to fast neutrons
Polyethylene	∼1.04	8.0	5.7	$$$	Useful for shielding neutrons. Particularly useful as borated polyethylene, with 5 % Boron by weight	
Paraffin	∼1.04	8.0	5.7	$$$	Useful for shielding neutrons	
Earth	∼1.5	Similar to concrete	3.9	Cheap	Low cost, easy to install	Density may be difficult to quantify
Wood	∼0.65	6 % by density	9.0	Cheap	Low cost, easy to fabricate	Large footprint in terms of barrier thickness

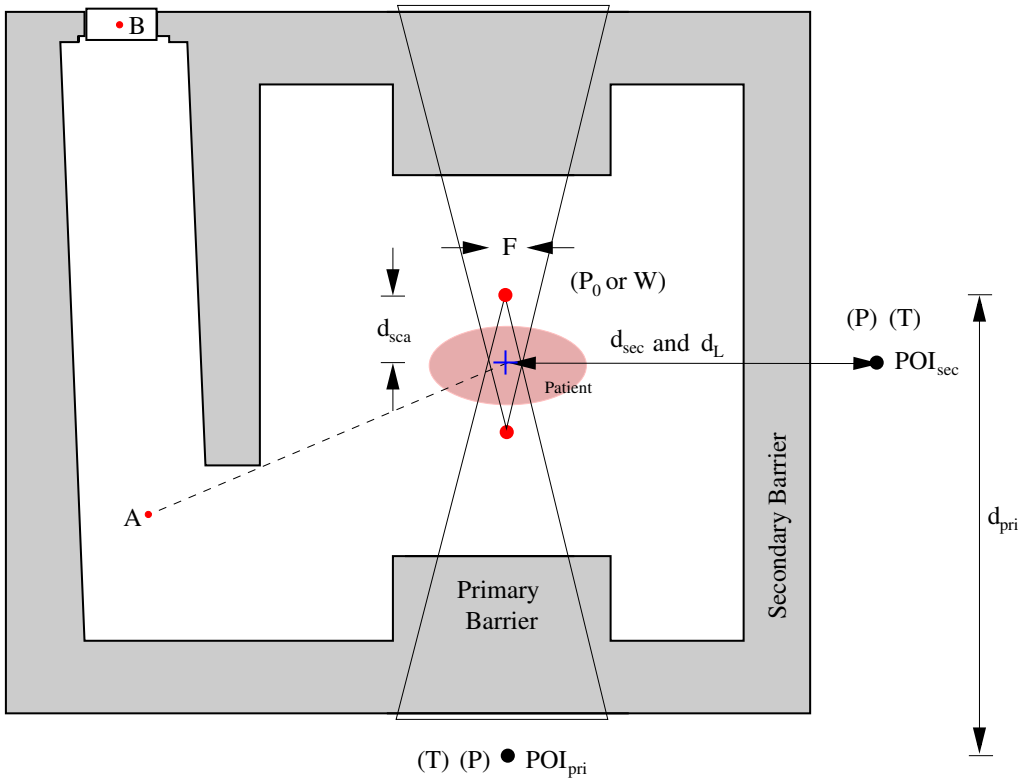


Figure 9.4: The geometry, distances and quantities involved in shielding calculations for primary and secondary barriers. Quantities are shown in parenthesis. P_0 and W correspond to the workload of the machine in the treatment room, with W in the same units as the design goal P. T is the occupancy factor of the room containing the POI. The letters A and B denote the positions of the inside and outside maze entrances respectively, of importance for neutron shielding considerations, as described in section **??**. The dashed line shows how position A is determined.

In the NCRP 151 calculation, the barrier attenuation factor B, by which the barrier must attenuate the dose produced inside the room to the desired level outside, is determined. The number of tenth value layers (TVLs) of barrier material required to provide B is then calculated. Finally, the barrier thickness is determined using the number of TVLs and appropriate TVL tables that are provided in the appendix of NCRP 151.

9.5.1 Determination of the Barrier Attenuation Factor B

The steps involved in the calculation of a barrier attenuation factor B are described here. The overall steps are the same for both the primary and secondary barriers but with important differences in the details, as described in section 9.7 below.

Step 1: To ensure compatibility of the units used in the shielding calculation, it is first necessary to convert the unit of dose rate used to describe the radiation source (linac or teletherapy machine) to the unit used to describe the design goal P. Typically, gray per year (Gy/yr) is the unit of choice for P, since it corresponds to the regulatory limits.

The workload W of a radiation therapy machine is a measure of the average dose it produces in water at its isocenter over a specified period of time. As such, the known or projected annual workload (in Gy/yr) is the quantity most appropriate for shielding calculations. A realistic and accurate estimate of W is vital for proper shielding design. It is a quantity that must be estimated by the designer from patient treatment projections. Unlike most of the other parameters encountered in a shielding calculation, W is specific to the treatment room involved and its value is neither prescribed nor recommended by the NCRP or by the regulatory authorities.

Step 2: A highly conservative estimate for B may be calculated by scaling W (i.e., the dose rate within the room) to the desired dose rate at the POI outside the room P, with the inverse-square law accounted for by a d^2 factor, where d is the distance from the source to the POI, as shown in Figure 9.4):

$$B = \frac{Pd^2}{W} \tag{9.1}$$

Location	Occupancy Factor (T)
Full occupancy areas (areas occupied full-time by an individual), e.g., administrative or clerical offices; treatment planning areas, treatment control rooms, nurse stations, receptionist areas, attended waiting rooms, occupied space in nearby building	1
Adjacent treatment room, patient examination room adjacent to shielded vault	1/2
Corridors, employee lounges, staff rest rooms	1/5
Treatment vault doors	1/8
Public toilets, unattended vending rooms, storage areas, outdoor areas with seating, unattended waiting rooms, patient holding areas, attics, janitors' closets	1/20
Outdoor areas with only transient pedestrian or vehicular traffic, unattended parking lots, vehicular drop off areas (unattended), stairways, unattended elevators	1/40

Table 9.2: NCRP 151 suggested occupancy factors.

Step 3: To determine a more realistic value for B (i.e., one that does not grossly over-shield), the room occupancy T at the POI and the usage U of the barrier must be considered. The resulting expression for B is thus:

$$B = \frac{Pd^2}{WUT} \tag{9.2}$$

The occupancy factor T is a scale factor that accounts for the true amount of time during which the room containing the POI is actually occupied over the calculation period. For example, if the POI is the control room, it can be assumed to have full occupancy ($T = 1$), since the radiation therapist is always at the console while the beam is on. If, however, the POI is in an adjacent room that is seldom occupied, it should have an occupancy factor less than 1.0. The list of room occupancy factors recommended by the NCRP are listed in Table 9.2.

The use factor U accounts for the fact that the radiation beam may be directed at the target volume within the patient from multiple directions by rotating the gantry. If the primary beam were only to strike the single primary

barrier between the source and the POI, that barrier's use factor would be 1.0. However, this is never the case, and the use factors of the four walls (two vertical walls plus the ceiling and floor) typically average out to 0.25 each. In certain situations, for example in treatment rooms used heavily for TBI (total body irradiation) or TSEI (total skin electron irradiation) treatments, the use factor of interest may be more or less than 0.25. The use factor is always 1.0 (and so is ignored) in secondary barrier calculations, since secondary radiation beams are assumed to be multidirectional.

At a first glance, equation 9.2 results in a value for B with units of m^2. However, the authors of NCRP 151 get around this by referencing all distances in the report to a distance of 1 m. Hence, the d^2 factor is divided by 1 m^2 and the resulting value for B is unitless.

9.5.2 Determination of the Number of TVLs and Barrier Thickness

With the barrier attenuation factor known, the corresponding number of TVLs n_{TVL} of the radiation beam in the barrier material is simply calculated using:

$$n_{\mathrm{TVL}} = log_{10}\left(\frac{1}{B}\right) \tag{9.3}$$

NCRP 151 recommends that, for primary barrier calculations, the first TVL TVL_1 should be considered separate to the remaining "equivalent" TVLs TVL_{e}. The authors reason that the hardening of the beam as it traverses the first TVL results in longer subsequent TVLs.

Finally, the required barrier thickness t_{barrier} is calculated as

$$t_{\mathrm{barrier}} = TVL_1 + (n-1)TVL_{\mathrm{e}} \tag{9.4}$$

NCRP 151 provides TVL thickness tables for ordinary concrete, lead and steel.

9.6 Primary Barrier Calculation

All the factors discussed above considered, the barrier attenuation factor for the primary beam is given by the equation:

$$B_{\text{pri}} = \frac{P d_{\text{pri}}^2}{WUT} \tag{9.5}$$

where a use factor less than 1.0 is typically employed. In calculating the primary barrier thickness, NCRP 151 makes the conservative assumption that the primary beam is unattenuated in traversing the patient (this is not the case in IPEM 75). Accordingly, when undertaking a post-construction radiation survey, a phantom should not be used in evaluating the primary barrier, if survey results are to be compared with NCRP 151 predictions.

9.7 Secondary Barrier Calculation

The secondary barrier must shield against both the leakage and scatter radiations. As can be seen in figure 2.6, at megavoltage energies the dominant photon interaction is the Compton effect. In the Compton interaction, the photon is not absorbed, rather it is scattered at an angle and with reduced energy. Both the leakage and scatter radiations undergo Compton scattering. The leakage beam scatters multiple times, and thus looses significant energy, while traversing the dense material (typically lead) in the head of the radiation generator. The scatter radiation, by definition, is scattered at least once before reaching the secondary barrier.

Two main sources of scatter radiation are possible: (1) scatter radiation produced by scattering of the primary beam within the patient, and (2) scatter radiation produced by scattering of the primary and patient scatter radiations with the walls and fixtures of the treatment room. A secondary barrier with thickness adequate to attenuate: (1) will generally be sufficient for (2), since

multiple scattering interactions reduce photon energy and penetrability significantly.

Barrier thicknesses are calculated for the leakage and patient-scatter beams independently due to the difference in energy between them. Leakage radiation, having undergone multiple scattering within the head of the treatment machine, is typically of a much lower energy than the patient-scatter radiation, which is scattered less often before reaching the walls.

As mentioned previously, a use factor of unity is used in secondary barrier calculations, since the secondary radiations are emitted in all directions.

Shielding Calculation for Leakage Radiation

IEC (International Electrotechnical Commission) regulations stipulate that the leakage from the head of a radiation-therapy unit must not exceed an average of 0.1 % and a maximum of 0.2 % of the primary beam over a 2 m radius measured from the beam central axis in the plane of the patient (IEC Publication 601-2-1, 1981). NCRP 151 takes a conservative approach and simply assumes that the leakage is equal to 0.1 % of the primary beam, even though the authors point out that manufacturers generally shield their machines to better than 0.1 %. The barrier attenuation factor for the leakage beam alone is thus given by the equation:

$$B_{\mathrm{L}} = \frac{P d_{\mathrm{L}}^2}{10^{-3} W_{\mathrm{L}} T}, \tag{9.6}$$

where d_{L} corresponds to the distance from the POI to the source of the leakage, which, as shown in Figure 9.4, generally averages to the isocenter. The leakage workload W_{L} may be assumed to be equal to the normal workload W except for clinical practice that involves a large component of IMRT—in IMRT

treatments, small field sizes are used and more "beam-on" time (larger leakage workload) is required to produce the same absorbed dose at the isocenter. Leakage-specific TVL tables are provided in the appendix of NCRP 151.

Shielding Calculation for Patient Scattered Radiation

The barrier attenuation equation used for patient scattered radiation is

$$B_{\mathrm{ps}} = \frac{P}{\alpha W T} d_{\mathrm{sca}}^2 d_{\mathrm{sec}}^2 \frac{400}{F}, \tag{9.7}$$

where (with reference to Figure 9.4):

- α is the scatter fraction or the fraction of the primary beam absorbed dose that scatters at a particular angle. Scatter fraction tables are provided in the appendix of NCRP 151.
- d_{sca} is the distance from the primary radiation source to the scattering material (the patient). Usually taken as the SAD of the machine.
- d_{sec} is the distance from the scattering point to the POI.
- F is the treatment field area (in cm^2) at the isocenter. The factor of 400 accounts for the fact that the scatter fractions provided in NCRP 151 are normalized to those measured for a 20 cm $\times$ 20 cm field size.

The scattering angle does not directly enter into the barrier attenuation equation but is considered instead in the scatter fraction lookup-table.

The Two Source Rule

Using the calculated barrier attenuation factors B_{L} and B_{ps}, for leakage and patient-scatter, respectively, the final required thickness of the shielding material can be determined. The barrier thicknesses required for each are compared as though there were two separate sources in the treatment room. If they are about the same (less than 1 TVL difference), then their combined dose at the POI should be about double their individual values. A half value layer (HVL)

is, therefore, added to the greater of the two to reduce the combined dose by one half. If the thicknesses differ significantly (by 1 TVL or more), the larger thickness will provide adequate shielding for both.

9.8 Maze and Door Calculations

As mentioned previously, the door of a treatment room controls access to the room while simultaneously providing shielding. One could employ a direct shielded door made from lead or steel and simply use the NCRP secondary barrier calculation methodology to determine its thickness. While adequate at low energies, such a direct shielded door would become prohibitively impractical at high energies where the secondary radiation is more penetrating. Above 10 MV, photoneutrons and neutron-capture gamma rays are an additional hazard that require further shielding. The solution is a maze barrier to attenuate the secondary radiation before it reaches the door. Figure 9.3 shows the geometry of a maze wall.

NCRP 151 provides a recipe to calculate the reduced secondary dose at the treatment room door when a maze is used. With regard to photon dose, the calculation method is the same regardless of energy; the energy being accounted for in the tabulated data provided. At high energies, however, the photoneutron and neutron-capture gamma-ray dose must be added to the photon dose.

9.8.1 Door Design

The thickness of the door required for a radiation therapy room may be calculated once the dose at the door has been established. For low-energy accelerator rooms, the door typically comprises a lead barrier encased in steel for rigidity. The transmission factor for the lead is calculated by dividing the

shielding design goal P for the area outside the door by the photon dose determined just inside it. The design goal may be modulated by the 1/8 occupancy factor, as per Table 9.2.

Doors used for high-energy accelerator rooms must shield against the photon dose together with the neutron and neutron-capture gamma-ray doses. A door comprising a sandwich of lead, borated-polyethylene (BPE) and lead, encased in steel is typically employed. The inner layer of lead serves to moderate fast neutrons from the treatment room by inelastic scattering, making the central layer of BPE more effective at thermalizing and absorbing them.[3] The outer layer of lead serves to attenuate the gamma rays that result from neutron capture by the boron. Once transmission factors have been established, both for lead and BPE, the required thicknesses of each may be determined from tables provided in the appendix of NCRP 151.

9.9 Radiation Shielding Evaluation

Following installation of a new radiation therapy device, an evaluation of the adequacy of the treatment room shielding is necessary before routine operation may begin. The objective of the evaluation is to determine if the design goal has been achieved and to search for possible radiation leaks. Physical inspection of the shielding and a radiation survey behind all barriers are required.

The purpose of a physical inspection is to ensure that: (a) the shielding is constructed as designed and (b) no regions of potential radiation leakage (e.g.,

[3] Polyethylene (the polymer found in plastic bags) is hydrogen rich and serves as a good material for the moderation (slowing down) of fast neutrons to thermal energies through elastic scattering. Boron-10 is very effective at absorbing thermal neutrons through neutron capture.

misplaced conduits) exist. An evaluation of the various interlocks, warning lights and signs that provide non-shielding protection is also required.

The radiation survey comprises comprehensive measurements of the radiation levels outside each barrier for all operating modes of the radiation generator. Each measured radiation level should be compared with the expected level determined from the shielding calculations used in the design.

In determining an expected radiation level, an inverse shielding calculation is essentially performed. The calculation begins with the known barrier thickness, from which the number of tenth value layers, and hence the expected attenuation factor, of the radiation beam in the shielding material may be calculated. Since the physical measurement is instantaneous, usage and occupancy factors of unity are used and the machine's workload is replaced by its instantaneous dose rate.

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